NUCLEAR SAFETY

A REVIEW OF RECENT DEVELOPMENTS

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foreword

This quarterly journal is a Technical Progress Review prepared by the Oak Ridge National Laboratory at the request of the Division of Information Services, U. S. Atomic Energy Commission. This Review is intended to assist those interested in keeping abreast of significant developments in the field of nuclear safety. *Nuclear Safety* is not a comprehensive abstract of all literature published in this field during a given quarter; rather it is a mechanism for presenting concise reviews of selected subjects as prevailing interest and available information warrant.

Coverage of the Review is limited to topics relevant to the analysis and control of hazards associated with nuclear reactors, operations involving fissionable materials, and products of nuclear fission. Primary emphasis is on safety in reactor design, construction, and operation; however, safety considerations in reactor fuel fabrication, spent-fuel processing, nuclear waste disposal, and related operations are also treated. Safety in the use of radioisotopes in industry, medicine, and research is excluded, as are most topics considered in the province of health physics. Even with these exclusions, nuclear safety overlaps such diverse fields as nuclear physics, solid-state physics, mechanics, chemistry, meteorology, geology, seismology, metallurgy, law, and nearly all branches of engineering. The authors will therefore review material from these fields which, in their opinion, has a direct bearing on nuclear safety.

All incoming literature (including reports, books, American and foreign technical journals, and transactions) is examined for subjects within our area of interest. This material is collected, grouped, and reviewed by experts. Interpretations, where given, represent the opinions of the editors, who are employed by the Oak Ridge National Laboratory. Readers are urged to consult the references to original work for more complete information. Many members of the Oak Ridge National Laboratory staff wrote review material, reviewed manuscripts, or otherwise contributed to this publication. Their contributions are gratefully acknowledged.

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Insurance: An Equalizer of Risk

By REUEL C. STRATTON*

Historically, the story of insurance is an interesting one. Some historians believe insurance has existed well over 6000 years, admitting that the early forms and methods were crude. It is definitely known that about 4000 B.C. the Babylonians had some means for transferring the chance of loss and that transit insurance existed by 2500 B.C.; loan contracts existed also. After 1600 B.C. the Phoenicians devised methods to insure shipments under the new conditions of water transportation. The Greeks quickly followed since they traded with the Phoenicians and thus learned their methods. About 900 B.C. the Rhodians, after acquiring knowledge of insurance from the Phoenicians, developed it further and laid down what is known as the Rhodian Law. This law still exists as the basis of general averages in contracts of marine insurance.

Although history is vague, fire insurance existed in China prior to 600 B.C., and shortly thereafter history recounts legislation upon the subject in both Greek and Roman laws.[1,2] Ways and means of distributing loss developed rather slowly until 1666 A.D., following the great fire of London. The first English fire insurance policy on record is dated 1686. The industry progressed in this country, with the first record being fire insurance policies issued in 1746 through a short-lived project in Charleston, S. C. In 1752 Benjamin Franklin established what is considered to be the first American fire insurance company.

Third-party liability insurance became established with the first employer's liability insurance written in England in the early 1880's, and workmen's compensation insurance was an outgrowth of employer's and common-law liability. In 1911 the Travelers Insurance Company issued the first workmen's compensation policy in the United States to an industrial concern which is still a policyholder.

Life insurance was recognized in 1574 when Queen Elizabeth granted the right "to make insurance policies" to Richard Chandler, who established a Chamber of Insurance. The earliest life insurance policy recorded was issued in 1583. In 1587 the first court case concerning life insurance was heard. The first known attempt at actuarial science was made in 1654 through a study of the experience of insurance

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companies by Pascal, who, after experimenting with hundreds of turns of roulette wheels, outlined the theory of probability which is now recognized as a mathematical principle and which is vital to actuarial science. In 1664, John Graunt computed the first probability table of life expectancies. Later, in 1693, Edmund Halley published the Breslau Table, which was the first mortality table based upon scientific principles. In 1705, the first life insurance company to be established on sound principles of mortality was chartered in England by Queen Anne. Other companies soon followed, and the first American life insurance company was established in 1812 as the Pennsylvania Company for the Insurance of Lives. The Panic of 1873 tested the soundness of the legal reserve principle in life insurance and the importance of diversified investments and assets.

It is thus evident that the entire insurance industry grew up piecemeal. The need for minimizing the chance of major economic loss became gradually apparent. Industrial and individual risks came to notice in many ways, and, as each chance of loss manifested itself, an insurance coverage was developed to eliminate or reduce it. From a simple beginning, probably more then 6000 years ago, the insurance industry today writes more than 400 different forms of coverage. In fact, the basis for the entire science of insurance was laid down by the ancient Babylonians, Phoenicians, Rhodians, Greeks, and Romans in a period which ended approximately 1000 B.C. Industry today could not survive without insurance. The operators of a large or small business or an industrial establishment must rely upon this commodity if they hope for a successful enterprise.

The average purchaser of insurance probably does not realize what has gone into the development of insurance to satisfy his need. Insurance has followed every endeavor of man, developing ways and means for protecting his investments or interests. It provides a way to ease his legal responsibility. The availability today of the many forms of insurance which apply to industrial activities has produced a feeling that the purchase of an insurance contract is, to a certain extent, comparable to the purchase of a can of fruit from a chain-store shelf; such is not the case. Insurance is individually underwritten, and the purchaser must qualify to obtain his coverage. The relationship between an insuring company and the purchaser of the insurance is, and must be, governed not only by the terms of the insurance policy but by the principles of law applicable to its interpretation. Basically, the policy is a contract; it constitutes a promise by the insurer, in consideration for a premium, to indemnify the insured for losses arising from certain specified events. An insurable interest is a prerequisite to a valid contract. Although the agreement may be speculative in a certain sense and the protection may be provided for a comparatively small sum, with one party assuming the risk of incurring losses in greater or lesser amount, the contract does carry with it the expectation that the insured will exercise every necessary effort to close the avenues toward known losses. It also carries the expectation that the insured will foresee, in so far as possible and with the assistance of the insurer, possible future exposures and eliminate them. Without the requirement that there must be an insurable interest, an indemnifying contract would be a simple wager. Instead of this, the premium payments required as a fee for protection are presently based upon the accumulated history of many losses incurred and are carefully calculated from actuarial data that take into consideration not only the probability of a loss but also the possible total extent of such losses.

There is a form of insurance coverage available for each need for protection cited. The coverage can be tailored to fit the exposure or the value involved.

Coverage by Insurance

There are certain responsibilities which must be assumed by the operator of any installation. These responsibilities are as follows:

1. The protection of the employees of the owner or operator.

2. The protection of the persons and the property of those who may be considered as being the public.

3. The protection of the investment.

Human experience has shown it to be commonplace that every innovation designed to improve the status of mankind has been attended by drawbacks which were not foreseen at its commencement. One example is the substitution of

railways for the older and more primitive methods of travel, although the immediate advantages which were obtained from a quicker means of transportation were many and apparent. The same premise applies to the introduction of electrical energy, automobiles, and, later, air transportation for use by the public.

Each innovation has brought forth at a later date new laws fixing the responsibility for the operation of the devices and requiring compliance with certain safety specifications if continuance of the device was to be permitted. The legal limitations placed upon industrial operations, as well as the laws concerning compensation and "third-party liability," are the aftermath of failure to design or operate devices in such manner as might be considered "in perfect safety."

In the middle of the 19th century it was found that the English common law, together with the Fatal Accidents Act, failed to provide proper compensation for cases in which negligence could not be established. Such findings have gone through a continuous evolution to the present status of such legal requirements. In fact, contributory negligence or assumption of risk because of peculiar hazards has little value today in limiting the liability of the owner and operator. There is a tendency to act in full legal belief that the development of any new project requires that the designer and developer assume complete responsibility for any incident which may occur and which may create a tort or damage upon any employee or upon the person or the property of the public.

1. *Protection of Employees.* Individuals who are alert to the future desire to protect themselves against unfortunate incidents through the establishment of a personal estate. The simplest means is through the many and varied forms of personal life and accident coverages which have been made available by many insurance companies.

Practically all states now make legal provision for the application of employers' liability or workmen's compensation. There is little difference between the two, except that under the common-law or employers' liability system an injured employee must generally show that he received injury under circumstances that would impose a legal responsibility on the employer; whereas the compensation system recognizes that industry should bear at least a proportional share of the financial losses caused by industrial accidents. The "safety first" movement in the United States and Canada is so well known today that it is unnecessary to discuss it. However, industrial accidents do have an effect upon public acceptance of projects. In addition to adherence to the legal requirements of protection of the employee by the employer, modern industrialists make certain that their plants follow certain principles, such as the following:

1. Careful selection of employees for specific jobs.
2. Establishment and enforcement of proper operating and safety rules.
3. Provision of a safe place in which to work.
4. Provision of proper safety equipment.
5. Education and training of employees in the control of the hazards of their duties.

During the last half century the effect of exposure to toxic materials within industry has received attention comparable to the prevention of injuries. Today most state jurisdictions recognize occupational disease and provide compensation for it. The physical effect of exposure to radiation is considered by most authorities to be an occupational disease.[3-5] Therefore the user of nuclear materials must assume this responsibility toward his personnel and provide adequate protective and control procedures.

2. *Protection of the Public.* In the relationship of an industrial operation to the so-called public, the law of tort or the law of negligence applies generally. Some authorities state that the law of negligence[6] does not prohibit the creation of risks to others. It does, however, consider the unreasonableness of the risk and contends that, although it is impossible in the business of life to avoid completely the risk of injury to oneself or to others, the law in itself does not forbid such but, in effect, does require that the risk be not unreasonably great.

In present-day decisions the courts have recognized no difference in common-law requirements with respect to situations containing various degrees of danger. The courts have decided that the precaution required must be appropriate to the risk and that the creation of an unreasonable risk to others is actual negligence.[7] Therefore it appears that the increased hazard from the radioactive products of a reactor, a chemical separations plant, or any other plant handling radioactive materials would be considered the creation of an unreasonable risk, for

which the owner and operator would be held responsible and liable for losses encountered.

Although an industrial activity may create some danger to others, the risk may not be regarded as an unreasonable one if the activity is of general social utility. The operator must take all reasonable precaution to avoid harm to others, but even so it has been ruled that the operator of a particularly socially desirable enterprise, which in itself contains the possibility to do harm, must assume the peril of compensating for any actual losses.

In a prominent legal decision[7] made many years ago, which was indicative of the law of negligence and the responsibility of an operator, it was stated by the courts: "We think that the true rule of law is that the person who, for his own purposes, brings on his land and collects and keeps there anything likely to do mischief if it escapes must keep it at his peril; and, if he does not do so, he is prima facie answerable for all the damages which are the natural consequence of its escape." It appears that this decision applies not only to the possibility of bodily injury but also to damage to property owned by others.

Thus many legal decisions show that the owner-operator of any enterprise (this could include any user or processor of nuclear material) must assume the responsibility cited, guard against it, and provide means for the just disposition of any substantiated and proven claims.

3. *Protection of Investment.* Responsibility of the operator for protection of the investment is self-evident. It is always necessary to protect one's capital, and this responsibility enters not only into the initial design of facilities and laboratories but also the operating procedures for use of the facilities. The designs of equipment and vessels are, of course, included. It is essential that all apparatus be properly engineered and developed for the purpose intended and that such equipment be located in accordance with established safety procedures. It is important that the controls and instrumentation in large and valuable chemical processes be such that continuity of effective production is established. Frequent unscheduled outages, unnecessary dismantling, and unwarranted catastrophes are costly. The too frequent dismantling of equipment for inspection or replacement and the cleaning of such equipment at arbitrary intervals could unnecessarily sacrifice production and increase maintenance cost.

Where pressure vessels are used, it is essential that all vessels be designed and built according to established codes, such as those of the American Society of Mechanical Engineers (ASME).[8] This generally means meeting the requirements of a specific state jurisdiction, which, in addition, may include compliance with the requirements of the American petroleum industry.[9]

Plant construction should always include proper fire protection, and adequate fire-fighting equipment should be available.[10] Proper waste-disposal facilities are essential. If in research or in the development of new processes it becomes absolutely essential to deviate from the usual safety precautions in design and construction, every effort must be made to assure that good engineering practices are followed in both design calculations and construction specifications.

Insurance in the Nuclear Energy Field

Personal Life Insurance for Atomic Industry Employees. In the early days of the atomic industry, the underwriting of individual life risks for persons who worked within restricted areas was virtually impossible. Life underwriters do not like to be confronted with applicants who are substantially "mystery men." This resulted in the arbitrary assignment of mortality ratings or additional rates when the applicant insisted that he be permitted to purchase insurance in some form. Even following the cessation of war activities and the emergence of the U. S. Atomic Energy Commission (AEC) to replace the Manhattan Engineer District, life underwriting was still doubtful. However, the Home Office Life Underwriters Association led the way by permitting the development of a coding system, acceptable to the AEC, which so penetrated security that, immediately, more than 97 per cent of all persons working in AEC plants were able to purchase their desired coverage as if they were not working in a plant subject to radiation. Since then, through continuous resurveys and the application of codings, together with the development of a reasonable quantity of actuarial experience, underwriting requirements have been

lessened until, today, almost all persons working in AEC installations may obtain their insurance exactly as if they were working in an ordinary industrial plant. In addition, a few years ago the accident and health insurance associations adopted similar codings and underwriting attitudes, and apparently they are concurring with this underwriting leniency. Thus the life, accident, and health companies have carried forward a program of evaluating the possible additional risk created through the introduction of nuclear energy into large plants. They have, during the past years, continuously collected statistical data and, in a forward-looking manner, have been able to absorb the exposure in a normal manner at a normal cost. There is no apparent reason why this approach should not continue.

Workmen's Compensation and Employers' Liability. This coverage is required by law in most state jurisdictions. When the law does not list a specific occupational disease within the Workmen's Compensation Act, such exposure may be covered under Employers' Liability by policy endorsement.

Third-party Liability. The Atomic Energy Act of 1954 gave private industry an opportunity to enter the field of nuclear energy. This placed exposures created by industrial participation in the nuclear energy field within the responsibility of the owner to make certain that all precautions appropriate to the risk were taken and that no operation created an unreasonable risk to others. The catastrophe features of explosive plants and large chemical plants were well known within the casualty insurance industry, and such knowledge permitted a reasonable gauge of the possibility and probability of serious accidents and injuries in the nuclear energy field. However, within the nuclear energy field, there was no factual experience available; there was only the admission that here, within one installation, were the means and the material which had a potential ability to do great damage to persons or to property. Such damage could easily foster claims of large proportion.

Physical Property Damage. The majority of nuclear energy installations, either large or small, could be carried under conventional physical-property-damage insurance coverages, except for one particular item; this item is radioactive contamination. Up to this time, no means had been developed to insure against the cost of the decontamination of property affected by radioactive material. Losses from such contamination could become great because of the inability to approach the area in which the loss occurred either to fight a fire, if such existed, or to decontaminate following a fire or other episode. When considering physical-property-damage coverage, it becomes apparent that a total loss could occur without a plant or its objects being visibly damaged, the loss simply being created by an inability to salvage or to reconstruct as a result of the nuclear exposure.

Availability and Cost of Insurance

Although it is admitted that all usual and normal nuclear operations can probably be handled satisfactorily insurancewise, the foreseeable substantial magnitude of possible loss which may be developed from reactor installations requires that the insurance industry proceed slowly. There has been little statistical experience developed to indicate, to any degree of satisfaction, either the possibility or the probability of a catastrophe. Few people are so presumptuous as to prophesy that a severe episode with a nuclear reactor cannot happen; and few, if any, will attempt to estimate the probability. It must be admitted that, to date, no episode of any great magnitude has occurred in this country. Because of this lack of loss experience, rate-making becomes difficult. The insurance industry must be prepared to face the fact that some persons exposed to radiation may not show evidence of the exposure for years, thus introducing the question of deferred liability. This is bothersome under a workmen's compensation law wherein personnel exposure is changed from its usual industrial complexion, since all the employees at the reactor installation may be exposed in a single event. This is a situation most unusual in present-day industry. Furthermore, such an event could endanger employees in neighboring plants, and there might be questions as to whether or not they would be compensation or public liability cases.

The casualty insurance industry has been diligently studying the situation. After the introduction of radioisotopes into industry for control, research, and medical purposes, the Joint Casualty Committee spent approximately one year reviewing data, visiting plants, and conferring with various state authorities and the

AEC. As a result of this study,* the committee concluded that the additional exposure created through the general use of radioisotopes in industry, research, and medical diagnosis or therapy could well be absorbed within normal insurance classifications and rates without additional underwriting or rate requirements. This decision has held, in general, since then.

The demand by industry for third-party liability policies with exceptionally high limits, and for physical-property-damage coverage of high-value plants subject to large losses through a nuclear episode or subsequent radioactive contamination, led the insurance industry to form three associations. These are the Nuclear Energy Liability Insurance Association (stock), Nuclear Energy Property Insurance Association (stock), and the Mutual Atomic Energy Reinsurance Pool (a mutual organization).†

Each of these organizations consists of a large number of companies banded together in such a way as to provide the desired capacity to assume risks heretofore not underwritten by individual companies or by groups of companies. Liaison between the "pools" permits approximately $60 million for liability, including foreign commitments. For physical-property-damage coverage, about $65 million is available.

The special nuclear liability policy now available will not be described here, except to say that it permits offering a very broad coverage of the nuclear energy hazard. All other liability must be covered by conventional means. Specific rating formulas have been developed which are acceptable to all organizations and to the majority of rating control organizations. The rating for each risk is developed individually by a Joint Rating Committee after an inspection by a Joint Engineering Subcommittee; the rating committee and the engineering subcommittee are selected from the pools.‡

Rating procedures established for nuclear liability policies have followed, on an individual basis, the generally accepted liability rating patterns. A typical quotation for a reactor installation of 630-Mw(t) capacity is $120,000 per year for $5-million limits. Quotations on various nuclear facilities to $60-million limits have ranged from $30,500 to $364,000. For small package type research reactors, the quotations run from $1500 per year for $1-million limits to $3500 per year for $5-million limits. In addition, the product liability form or the shipper's and transporter's form of coverage is available.

Although at first glance such quotations may appear to be sizable sums of money, close analysis will show that the figures quoted are far below the cost of obtaining loans of comparable dollar amounts on a stand-by basis only from large financial institutions. In addition, the insurance industry has introduced a 10-year credit rating plan that was developed to permit a large retroactive premium return if public liability losses from nuclear hazards remain nonexistent or low during the same 10-year period. This plan permits the development of a reserve for losses during the early influx of the development of nuclear hazards and then a gradual return of the reserve over and above what experience shows to be necessary.

With respect to physical-property-damage coverage, an all-risk package policy that combines coverage for perils usually underwritten separately has been developed for the first time. Basically this was done because the insurance industry believed that the adjustment of losses would be extremely difficult if coverages for various perils were insured separately. For example, if a large pressurized-water or boiling-water reactor exploded, either due to overpressure or to a nuclear incident, it would be relatively impossible to tell which occurred first; therefore, the all-risk policy was selected. This also permitted the introduction of coverage for any loss sustained due to on-site contamination, including the actual loss of property or the cost of removal of contamination.

The rating for such all-risk physical-damage policies has been simplified. For the usual perils such as fire, extended coverage, vandalism, and malicious mischief, the inland marine and other comparable forms of coverage carry the conventional rates for the jurisdictional area in which the risk is located. Boilers and

*Prepared by the Joint Casualty Committee, composed of six large insurance companies now absorbed by the Nuclear Energy Liability Insurance Association and the Mutual Atomic Energy Reinsurance Pool.

†Nuclear Energy Liability Insurance Association, 60 John Street, New York 38, N. Y.; Nuclear Energy Property Insurance Association, 85 John Street, New York 38, N. Y.; Mutual Atomic Energy Reinsurance Pool, 919 North Michigan Avenue, Chicago 11, Ill.

‡The committees are selected from members representing the Nuclear Energy Liability Insurance Association and the Mutual Atomic Energy Reinsurance Pool.

machinery are handled on a conventional basis, except that the premium is calculated as a value charge. The all-risk feature is of minor importance. This leaves one item, namely, the so-called "nuclear rate increment," which heretofore has not been available. To develop the cost of this nuclear coverage feature, special rating procedures acceptable to the companies and to rating jurisdictions have been developed. Such procedures (1) evaluate the varying degrees of loss potential from a nuclear energy standpoint upon each individual risk and (2) hand tailor the rate to the exposure. Although only on-site contamination is covered within this all-risk policy, it is possible to obtain, by endorsement, contamination coverage upon property of the same ownership located contiguous to, or at a distance from, the original location.

Insurance is a valuable tool with which industry may cushion the effect of severe losses. The introduction of nuclear energy into the insurance field by creating a demand for exotic insurance coverage stimulated the insurance industry to provide the forms of coverage which are now available and which do provide adequate protection to any and all nuclear energy endeavors. Insurance is the most satisfactory way to equalize the risk. (Reuel C. Stratton)

ICRP Radiation Protection Recommendations

Since its formation in 1928, the International Commission on Radiological Protection (ICRP) has been perhaps the most influential organization concerned with setting appropriate standards for exposure to ionizing radiation. Reporting only to the International Congress of Radiology, the ICRP has been a means of obtaining substantial uniformity among the many national codes dealing with protection against radiation, as well as a potent force acting to ensure that protective measures are adequate. Each successive report of the Commission has been marked by a more extended scope and increased detail; thus the present publication[11] of the main commission of the ICRP deals, in principle, with all the major aspects of radiation protection. The reports of its various subcommittees are planned to apply the general principles of the recommendations of the main commission to specific areas such as internal dose, heavy particles, and waste disposal. In this Review only major differences or changes of emphasis of this report as compared with earlier reports will be mentioned.

The 1958 report[11] gives much more attention to exposures not directly arising in the course of employment than the 1955 report.[12] The following exposure categories are defined:

1. Occupational exposure.
2. Exposure of special groups.
   - *a*. Adults who work in the vicinity of controlled areas but who are not themselves employed on work causing exposure to radiation.
   - *b*. Adults who enter controlled areas occasionally in the course of their duties but who are not regarded as radiation workers.
   - *c*. Members of the public living in the neighborhood of controlled areas.
3. Exposure of the population at large.
4. Medical exposure.

Recommended dose limits are given only for the groups under categories 1 and 2. Dose limits and practices are "suggested" rather than recommended for category 3, but it is likely that they may have equal influence with the public.

The recommended dose limits for groups 1 and 2 take into account many effects and conditions not cited explicitly in previous reports, e.g., accumulated dose or lifetime dose as opposed to dose rate (rem/week), accumulation of genetic effects in a population, group size, recognition of the need for some upper limit on the total dose from man-made radiation in a population, and a first attempt to delimit the contributions from various activities. The bases for the dose limits recommended or suggested are put forth in much greater detail than in any of the earlier reports, as would be expected in view of the vastly increased public interest in these matters.

The increase in detail and variety of the exposure categories is matched by the diversity of the limits which are recommended or suggested. An explicit limit is imposed on the accumulated dose, or lifetime dose, when the exposure involves the gonads, the lenses of the eyes, or the entire hematopoietic system. Thus the concept of irreparable damage or of a strictly linear dose-effect relation seems to be implied. For example, in the above-mentioned organs, the accumulated occupational dose is

limited to $5(N-18)$ rem, where N is the worker's age in years. This recommendation is not intended, however, as a mere necessary consequence of a dose-rate limit since the worker is still permitted to accumulate 3 rem in any period of 13 weeks or 12 rem in a year, provided his accumulated occupational dose is not in excess of the above limit. This new rule reduces the maximum accumulated dose to these organs to about one-third the previous theoretical maximum implied by the dose-rate limit of 0.3 rem/week. In practice, occupational exposures have seldom, if ever, persisted at such a level for long periods, and the present age of the atomic energy industry did not permit the accumulation of very large doses even at the former rates. Under the previous limit of 0.3 rem/week, only the medical practitioner and perhaps some experimentalists were likely to reach a 50-year occupational dose of $50 \times 50 \times 0.3$ rem = 750 rem. The new rule amounts to a recognition of the vastly increased number of workers routinely engaged in work where there is potential exposure and thus recognizes that atomic energy is an employment category of major proportions.

The detailed consideration of the special subgroups of group 2 gives increased emphasis to the general point of view that hazards peculiar to one industry or operation must not be extended to other groups without some diminution in the level of exposure and due concern for the constitution of the exposed groups. The ICRP has recommended that the exposure doses to these groups not exceed 0.3 of the recommended occupational limits for the same type of exposure in the case of subgroups 2*a* and 2*b* and that the reduction factor be 0.1 for subgroup 2*c*.

The values of suggested limits for exposure of the whole population are essentially a new development in health physics. These limits became necessary because of recognition that potentially, and to some extent actually, the entire population is exposed to man-made sources of radiation other than medical. Since effects of ionizing radiation, such as genetic mutations and sterility, may affect the future of the human race, as well as the lives of the individuals receiving the dose, new criteria and limits were needed for control of effects on the population as a whole. It is clear that a very small change in the mutation rate, although it would represent only a very slight increase of risk to any single individual, might, if extended to the population at large, have more serious consequences. Thus, instead of the damage to an individual or the distribution of damage at any given time, the primary concern is the total burden of genetic damage resulting from the propagation of radiation effects throughout the race. Consequently, all sources of ionizing radiation must be considered as contributing to this total burden. Accordingly, the report cited in reference 11 considers the contributions to the genetic burden from background radiation, medical and dental exposures, and other man-made sources.

Present indications are that medical exposures may deliver as much as 5 rem (to age 30 years) of genetic exposure in many countries where X rays are commonly used for diagnosis and therapy. There is some indication that this rate may be declining as practices and equipment are improved. The report[11] does not give any suggested limit for medical exposure but does indicate that the subject is under study and urges that such exposure be held to a minimum, consistent with sound medical practice.

The report[11] suggests that all other genetic exposure to man-made sources of ionizing radiation be limited to an additional 5 rem per 30 years. Since the additional 5 rem per 30 years must include a wide variety of exposures, the report gives some indication of a possible apportionment of such doses. The contributions from occupational exposure and from exposure of special groups will depend on the size of these groups, as well as on the recommended limits, and thus any such apportionment may have to be adjusted as circumstances vary from nation to nation or with the passage of time. Taking into account the present size of these groups and their expected growth in the next few years, the Commission suggests the following apportionment:

Occupational exposure	1.0 rem
Exposure of special groups	0.5 rem
Exposure of the population at large	2.0 rem
Reserve	1.5 rem

The reserve is primarily to allow for new developments and applications and to provide some margin of safety.

Somatic damage is even more difficult to assess so far as its probable impact on society is concerned. The report recognizes this but

also recognizes the need for some permissible level, if only for purposes of design and planning. Tentatively, it suggests that the somatic dose be limited to $\frac{1}{30}$ of the maximum permissible doses for occupational exposure.

Since the publication of the ICRP report, it has been questioned whether the limits for exposure of the entire population were intended as maximum limits on individual exposure or as limits on average exposure. A careful reading indicates that the latter interpretation is correct in most instances. The ICRP has recognized the need for a maximum limit on individual exposure and will issue shortly a supplement which will clarify this point. It will be specified that the presently indicated limits are for the average in the population, and the suggested maximum limits will be those recommended for subgroup 2*c*.

It is apparent that the problem of arriving at proper maximum levels of exposure to radiation is exceedingly complex. It is easy to criticize the bases for various positions since there are many factors affecting the results which are not precisely known at present. An excessive conservatism which insists on extremely low limits or a reckless acceptance of high limits which temporarily may be convenient are both to be condemned. An insistence on very low limits could seriously impede, or even prevent, the benefits that atomic energy has to offer to our civilization, and any serious consequences of a reckless disregard of proper concern for the protection of people would undoubtedly raise obstacles of major proportions to further development. The new report[11] evinces the careful consideration and *expertise* of the members of the ICRP, as well as their dedication to the task of making available to our civilization the benefits of radiation without incurring the hazards its reckless use might entail. (W. S. Snyder)

Nuclear Pressure and Containment Vessel Code

It is generally accepted that continued industrial progress in the atomic energy field in the United States requires the development of adequate and uniform nuclear safety standards,[13] but it is not always entirely clear who will generate these standards. The AEC, through its various divisions and the Advisory Committee on Reactor Safeguards, has ultimate responsibility for approval of the safety aspects of such systems, but it does not prepare specific detailed standards for commercial atomic energy installations. Instead, the AEC encourages recognized standards-writing bodies to develop such standards.[14] In some cases, existing standards-writing bodies have extended the scope of their work to include the nuclear field, but other organizations that specialize in nuclear problems are also working on the production of standards[13,15] (see page 13 of this issue for a summary of the efforts in the development of standards in the nuclear field).

The standards problem has been partly resolved for the components of pressurized power-producing systems by the fact that such components, nuclear or otherwise, generally fall within the scope of state or municipal laws and regulatory agencies[16–18] which, in turn, rely on the ASME Boiler and Pressure Vessel Code[19] for technical guidance. It is unlikely that other standards groups will function within the fairly well-defined limits established by the Boiler Code's relationship with the regulatory bodies.

ASME Boiler and Pressure Vessel Code

The ASME Boiler and Pressure Vessel Code was first adopted in 1915 to develop adequate and uniform boiler-construction rules for the use of the state and municipal regulatory bodies which were being established to reduce the many serious accidents and resultant loss of life and property that accompanied the continuing increases in pressures and temperatures of steam boilers. Since its original and successful adoption, the Code has been continually revised and enlarged to keep pace with subsequent developments in the steam power industry. It has also been expanded to include rules for unfired pressure vessel construction. Today it is the accepted standard for steam boilers in most states and major cities of the United States and the provinces of Canada.[20] Similarly, the Unfired Pressure Vessel Section of the Boiler Code has been adopted by regulatory bodies in perhaps half the states and major cities in the United States and by all the provinces of Canada;[20] it is also a generally accepted industrial standard for determining the insurability of such equipment, even when not required by law. Revised editions of the Code are issued at approximately three-year intervals. It is further kept up to date by supplements issued once or twice each

year and by "Case Interpretations" issued still more frequently. The Case Interpretations are intended to clarify the Code committee's intent in the application of Code rules to special design questions and also to authorize the use of newer materials in Code construction.

Code Interpretations for Nuclear Vessels

When American industry started plans a few years ago to build and operate privately financed nuclear power plants, it became apparent that many components of such installations would, in most instances, come within the scope of state or municipal boiler and pressure vessel laws and regulatory bodies and that the Boiler Code should consider the possibility of making provisions for the equipment in such plants. A special ASME committee on nuclear power was therefore established to advise the Code committee and its various subcommittees on matters pertaining to nuclear power which come within the scope of the Code.[21] About three years ago, the special committee determined that the ASME Boiler and Pressure Vessel Code contained most of the requirements properly applicable to vessels for nuclear service and that no comprehensive separate Code for such vessels was required at that time. It was decided to make maximum use of existing Code sections and to supplement them with Case Interpretations to cover the specific requirements of nuclear vessels. A series of Case Interpretations (1224 to 1226, 1228, 1234, and 1235) was then issued to accomplish this. More recently, the intent and wording of these original case interpretations have been incorporated into a series of Nuclear Case Interpretations designated by the suffix "N." Cases 1270N to 1274N are discussed below.[22]

Case 1270N, a "permissive" case, describes the general rules under which nuclear vessels can be built to Code requirements. It requires that such vessels comply with the applicable provisions of all other N-suffixed cases and with either the Power Boiler Section or the Unfired Pressure Vessel Section of the Code. It requires that circumferential and longitudinal welded joints be fully radiographed double-butt welds or the equivalent; it requires stress relief of reactor, primary, and containment vessels in all thicknesses; it states the limits of Code jurisdiction over piping external to the vessels; and it defines the various types of vessels used in nuclear installations, i.e., reactor vessels, primary vessels, secondary vessels, containment vessels, and intermediate containment vessels.

Case 1271N prescribes modifications of the requirements of other sections of the Code for safety and relief valves and pressure-indicating devices for nuclear vessels, eliminates the requirements for direct-reading pressure gauges in pressurized- and boiling-water reactors and for pressure-relief devices in containment vessels, and provides for the use of rupture disks in some locations.

Case 1274N makes special provision for the use of modified American Iron & Steel Institute (AISI) type 403 steel in certain nuclear pressure vessel applications.

Case 1272N covers the requirements for containment vessels. It waives some stress-relief requirements by (1) imposing low-temperature impact requirements on the vessel materials, (2) by requiring that frames for openings be preassembled into shell plate and be stress-relieved prior to installation, and (3) by applying additional inspection requirements to certain welds. It makes special provisions for intermediate containment vessels. It provides suitable inspection for two-stage construction of large containment vessels. It should be noted, however, that containment vessel construction meeting the requirements of Case 1272N is specifically limited to the fabrication of fine-grained, normalized-carbon-steel vessels designed for a minimum pressure of 15 psi and conventional Code stresses. Case 1272N also specifies the essential hydrostatic or pneumatic tests of Section VIII and substantially complete radiographic inspection of all principal pressure-containing welded joints. When conventional Code vessels are designed and built to similar requirements, they are generally intended to operate for extended times at design conditions, whereas full design pressure and temperature of a containment structure represent the maximum conditions to which it might be subjected not more than once, if at all, in the lifetime of the vessel. At the same time, the containment vessel does constitute the last safety barrier in the system and should be completely capable of withstanding this single most severe incident, should it occur. These significant differences from the service conditions for conventional pressure vessels, coupled with the accepted limitations of Code jurisdiction, have prompted many questions and

suggested modifications of the Case, and the entire problem of containment is under continuous review by the Code committee and other groups.[23–25]

Among the modifications that have been suggested are (1) increases in permissible design stresses, (2) changes in scope to include containment vessels designed to 5 psi or more, (3) optional use of concrete containment vessels, and (4) provisions for protection of the structure from ballistic missile damage in the event of failure of the contained system. The subject of design stresses as related to service requirements is under active investigation by a special ASME committee established to review the Code stress basis for all types of pressure vessels, not just nuclear containment structures. Significant selective modification of Code design stresses for a specific service, such as containment vessels, is unlikely except as a part of an over-all revision of the Code stress basis to include service conditions as determining criteria. Problems of allowable stresses, lower design pressures, missile protection, and use of nonmetallic structural materials have been considered by the subcommittee on containment of American Standards Association, Inc. (ASA) Committee N6 on Reactor Safety Standards. The subcommittee has prepared a draft copy of a proposed safety code for containment structures[25] which is now being reviewed by committee members. This proposed code, or some modification of it which accepts the Boiler Code where it is applicable but also provides for those situations which are outside the established scope of the Boiler Code, may well become standard for containment structures if the details of acceptance by regulatory agencies can be resolved.

Case 1273N covers special requirements for reactor vessels and primary vessels. It calls for compliance with basic design stress requirements of either the Power Boiler Section or the Unfired Pressure Vessel Section of the Boiler Code. It limits the combination of membrane stresses evaluated under conventional Code rules and steady-state thermal stresses to 1.5 times the Code design stress value. It establishes bolt design criteria and makes specific provisions for compensation for openings. It requires full-penetration, fully radiographed welds, including welds at nozzle attachments, and supplementary nondestructive testing when radiographing is not feasible. It makes provision for consideration of the application of cladding in design calculations. In some instances where there is, as yet, insufficient knowledge to permit setting minimum requirements on a sound basis rather than an arbitrary one, and in other instances where Code rules do not normally take the effects of service conditions into account in setting minimum requirements, the Case Interpretation places the responsibility directly on the designer to give "particular consideration" to these effects. For example, it requires that due regard be given to design details that might be affected by thermal stress or external pipe reaction. It requires that particular consideration be given to the deterioration that may result from corrosion, erosion, radiation effects, material instability, transient thermal stresses, mechanical shock, and vibratory loading. To ensure soundness and good practice, it requires that "special consideration" be given to materials, construction, inspection, supplementary methods of nondestructive testing, smoothness of welds, and other structural details.

Limitations of Existing Code

Case 1273N, added to the other Nuclear Case Interpretations and the applicable sections of the basic ASME Boiler and Pressure Vessel Code, is essentially equivalent to a nuclear pressure vessel code. It has limitations in that it requires "due consideration" instead of stating minimum standards in several instances. On the other hand, the Code is not intended to be a design handbook. The application of arbitrary but unsubstantiated m i n i m u m requirements might well prove either inadequate or unduly restrictive, and the due-consideration provision does place responsibility squarely where it belongs, i.e., on the responsible nuclear design organization. The absence of quantitative minimum standards emphasizes the areas warranting research effort to provide the appropriate criteria.

The more important limitations of the Boiler Code and its nuclear supplements as a comprehensive safety standard lie in the restrictions in scope resulting from the defined basic functions of the Code. The Code is largely limited to the pressure-containing capabilities of individual components and does not deal with integrated pressure-containing systems. The Code committee disclaims responsibility for considering

biological hazards from radiation. The Code does not make specific provisions for vessels operating at essentially atmospheric pressure, such as radiochemical and waste-disposal equipment. It is primarily a Code for new construction and does not necessarily make design provisions for circumstances in which normal periodic reinspection may be difficult or even impossible. The nuclear groups within the Boiler Code organization are aware of the areas outside the scope of the Code which should nevertheless be covered by a safety code, and they can be expected to take proper steps to encourage complete coverage by appropriate responsible groups. In the field of pressure-containing nuclear equipment, the preparation of appropriate and comprehensive nuclear safety codes will probably be performed by subcommittees and task groups responsible to ASA Committee N6 (reference 26), which is jointly sponsored by the ASME and the American Nuclear Society (ANS). The ANS has a standards committee that consists of a number of project groups which work, as circumstances require, on the instigation, preparation, or review of standards for the whole nuclear field.[26] ANS Project 7, Reactor Components, is specifically concerned with pressure-containing systems and their components. The corresponding ASA Subcommittee is N6-3, Reactor Fluid Systems and Fuels Within the Reactor (see the section on Organization for Standards in the Nuclear Field, page 13 of this issue).

The ASA Committee B31, Code for Pressure Piping, has an actively functioning advisory committee on nuclear piping[27] which makes recommendations to other piping code committees on nuclear problems. The Piping Code differs from the Boiler Code in that the Piping Code has not, for the most part, been adopted as mandatory by governmental regulatory bodies. In the absence of the need for mandatory minimum requirements, a number of details in the Piping Code are presented as optional provisions for adoption by mutual agreement between the seller and buyer, if they so desire. Various sections of the Piping Code, including the Power Piping Section, which will be basic in most nuclear power applications, are being reviewed and revised. The current version, B31.1-55, of the Power Piping Section is usually augmented with job specifications when applied to nuclear systems.

Materials specifications for nuclear service are generally prepared by the American Society for Testing Materials (ASTM). The ASTM is actively promoting the review and, where necessary, modification of its specifications to ensure their applicability to the nuclear field.[14] For materials to be used in pressure-containing equipment, the Boiler Code committee reviews the ASTM materials specifications to ensure their adequacy, sometimes makes changes in them, and then incorporates them into Section II, Materials, of the Code. If no ASTM specification is available for a material, as in the case of some proprietary alloys, the Code committee may authorize the use of the material by issuing a Case Interpretation. There is also a considerable number of federal or military materials specifications, some of which are comparable to ASTM specifications, employed in nuclear installations owned by the federal government.

Summary

This discussion of the availability and stages of development of codes, standards, and specifications for general use in the industrial power-reactor field has intentionally made little mention of the activities of the federal government. It should be noted that the AEC has a considerable regulatory responsibility for the approval and control of these installations, even when they are constructed according to existing Codes. Also, there are other federal agencies with regulatory responsibilities for privately owned nonnuclear pressure-containing equipment employed in interstate commerce, and these agencies make considerable use, in whole or in part, of codes such as the ASME Boiler and Pressure Vessel Code. They will presumably do this also for pressurized nuclear equipment within their jurisdiction. Although federally owned facilities are legally exempt from the Boiler Code, the Code is often used, sometimes in its entirety, sometimes in part, as a basis for design and procurement of nuclear components. Even when federal agencies develop specialized codes of their own for nuclear equipment, many of the provisions are interchangeable with the commonly accepted industrial standards.

Efforts in the field of nuclear standardization indicate recognition of a compelling urgency for the establishment of such standards, but progress in the development of the standards will be slow because valid standards can be adopted

only when they are based on adequate technical information, experience, sound engineering judgment, consensus acceptance by many divergent interests, and, perhaps, the solution of complicated jurisdictional problems.

Organization for Standards in the Nuclear Field

The case for standards is that they save money, manpower, materials, calendar time, and lives. The case for national standards no longer requires argument. It was won long ago largely from two dramatic examples: the victory over the horrors of frequent boiler explosions[16,17] and the industrial victory of mass production, both made possible by standardization. American business has been slow to realize the importance of international standards (an exception is the motion-picture industry) but is now attempting to make up for lost time in many fields.

In the nuclear field, many of the standards required are unique either in kind (e.g., radiation standards) or in degree (e.g., leaktightness). Because of the rapid expansion of theoretical and experimental data and operating experience, standards must incorporate as reasonable a combination of rigidity and flexibility as the individual considerations warrant so that the benefits of standardization may be derived without stifling progress by the establishment of unwarranted restrictions.

Among the international groups actively engaged in developing standards that either directly or indirectly affect nuclear safety are the International Organization for Standardization (ISO), the International Electrotechnical Commission (IEC), and the recently created Pan American Standards Committee (PASC). The ASA represents the United States in the ISO, the IEC, and the PASC.

In the United States, there are several hundred organizations more or less actively working in the standards field. The ASA is a federation of 118 organizations supported by about 2200 companies, and about 60 organizations are represented on its Nuclear Standards staff. Among the organizations active in this field in the United States are several federal government bodies, technical societies, business associations and organizations, public and private nonbusiness organizations (e.g., the National Safety Council), labor unions, and associations of public health officials.

Some standards affecting nuclear safety are prescribed by federal[28] and state[29] laws and regulations (notably, the ASME codes on containers and vessels). In a sense the specific requirements of federal regulations enacted by the executive branch of the government under the Atomic Energy Act may be called "standards," as may the various specific state laws in this field.

In addition to the AEC, which is the main contributor to federal regulations regarding safety in the use of nuclear energy, there are other federal bodies interested and active in the field. Two significant examples are (1) the Federal Radiation Council, which was established by executive order on Aug. 14, 1959, to advise the President on radiation matters and which was given statutory status Sept. 23, 1959, by Public Law 86-373 specifying the Secretary of Labor as a member, and (2) the National Advisory Committee on Radiation (NACOR), which was established Feb. 12, 1958, to advise the Surgeon General on further development of Public Health Service programs dealing with radiation from all sources.

The major American producers of published standards directly affecting safety in the use of nuclear energy are listed in Table I-1. The ASA emerges as the most generally accepted source of comprehensive nuclear standards. For convenience, Table I-1 lists only the organizations which have reported that they produce and publish standards in this field under their own names. [In the important field of reactor safety, ASA and ANS standards committees (ASA sectional committee N6 and ANS Standards Projects 2 to 8) have essentially the same responsibilities, although the responsibilities are somewhat differently divided between the subcommittees. Therefore, to avoid excessive duplication, the specific details of coverage are described only for the ANS committees in Table I-1.]

The organizations working directly with the ASA Nuclear Standards Board and sectional committees on the production of American standards in this field are listed in Table I-2. Some of these organizations also produce and publish standards in other fields under their own names.

The methods of production and promulgation of standards differ in some details in the dif-

(Text continues on page 20.)

Table I-1 LIST OF ORGANIZATIONS* PRODUCING STANDARDS DIRECTLY INVOLVING SAFETY IN THE USE OF NUCLEAR ENERGY

Ia

American Standards Association, Nuclear Standards Board, 70 East 45th Street, New York, H. G. Lamb, Secretary. This board has 33 organization members and two members at large. President: Morehead Patterson.

ASA Sectional Committee N1. *American Standard Glossary of Terms in Nuclear Science and Technology*. This committee has already published *Glossary of Terms* (N1.1, 1957). The committee has been dissolved, and Subcommittee N2-4 will handle all revisions and additions to the glossary.

ASA Sectional Committee N2. *General and Administrative Standards for Nuclear Energy*. Sponsor: Atomic Industrial Forum, Inc., 3 East 54th Street, New York 22, N. Y. This committee has 17 organization members. It studies standards, specifications, and methods of administration associated with the peaceful uses of nuclear energy, including color codes, symbols, nomenclature, qualifications of professionals, records and reporting systems and procedures, and accountability of materials. The following subcommittees have been established:

N2-1 Color Codes and Symbols
N2-2 Procedures for Industrial Exposure Records
N2-3 Qualifications of Nuclear Professionals
N2-4 Nuclear Terminology (also represents ASA on ISO Technical Committee 85-SC1: Terminology, Units, and Symbols)
N2-5 Model Atomic Energy Legislation

ASA Sectional Committee N3. *Nuclear Instrumentation*. Sponsor: Institute of Radio Engineers, 1 East 79th Street, New York 21, N. Y. This committee has 21 organization members. It represents ASA on IEC Technical Committee 45, Electrical Instrumentation. The committee studies standards, specifications, and methods of testing for instrumentation in the nuclear field, including instruments for personnel protection, reactor control, industrial processes, analysis and laboratory work, radiation calibration equipment, and components therefor. Subcommittees are as follows:

N3-1 Index of Nuclear Standardization Work (a report has been published: "Index of Nuclear Standardization Work" by R. F. Shea; this report is obtainable from L. G. Cumming, Technical Secretary, Institute of Radio Engineers, Inc., 1 East 79th Street, New York 21, N. Y.)
N3-2 Topical Structure

ASA Sectional Committee N4. *Electric Apparatus and Systems for the Nuclear Field*. Sponsors: American Institute of Electrical Engineers, 33 West 39th Street, New York 18, N. Y.; Electric Light and Power Group, consisting of Edison Electric Institute, 750 Third Avenue, New York 17, N. Y., and Association of Edison Illuminating Companies, 51 East 42nd Street, New York 17, N. Y.; and National Electric Manufacturers Association, 155 East 54th Street, New York 17, N. Y. This committee has 12 organization members. No subcommittees are listed by ASA. Standards for electrical power apparatus, systems, and controls used in the nuclear field are considered by this committee.

ASA Sectional Committee N5. *Chemical Engineering for the Nuclear Field*. Sponsor: American Institute of Chemical Engineers, 25 West 45th Street, New York, N. Y. This committee has 22 organization members. It considers standards, specifications, tolerances, and methods of testing for the chemical engineering aspects of the nuclear field, including the refining, processing, separation, purification, treatment, packaging, handling, and disposal of fuels, radioisotopes, industrial chemicals, pharmaceuticals with isotope tracers, and radioactive wastes (gases, liquids, and solids); the use of radioactive sources for tracers and for processing foods and other materials; the application and use of chemically resistant coatings; and the cleaning of contaminated equipment and facilities. Subcommittees are as follows:

N5-1 Fuel Manufacture and Fabrication
N5-2 Radioactive-waste Disposal
N5-3 Recovery of Irradiated Fuel
N5-4 Use and Handling of Radioisotopes and High-energy Radiation (also represents ASA on ISO Technical Committee 85-SC4, Radioisotopes)
N5-5 Packaging and Transportation of Radioactive Material

ASA Sectional Committee N6. *Reactor Safety*. Sponsors: American Nuclear Society, 86 East Ran-

*This list includes only those organizations which produce standards under their own name. Other organizations which contribute to the ASA standards work are listed in Table I-2.

Table I-1 LIST OF ORGANIZATIONS PRODUCING STANDARDS DIRECTLY INVOLVING SAFETY IN THE USE OF NUCLEAR ENERGY (CONTINUED)

dolph Street, Chicago 1, Ill., and American Society of Mechanical Engineers, 29 West 39th Street, New York 18, N. Y. This committee has 43 organization members. It represents ASA on ISO Technical Committee 85-SC3, Reactor Safety. It considers codes and standards concerned with the hazards involved in the design, location, construction, and operation of nuclear reactors and of potential critical assemblies. Subcommittees are as follows:

- N6-1 Site Evaluation*
- N6-2 Containment†
- N6-3 Fluid Systems of the Reactor and Fuel Within the Reactor
- N6-4 Reactor Dynamics and Control Requirements
- N6-5 Instrumentation and Execution of Control Requirements
- N6-6 Operation, Operator Qualifications, Inspection and Maintenance, and Records
- N6-7 Failure Probabilities and Maximum Credible Accidents
- N6-8 Fissionable Material Outside Reactors (normally subcritical systems)

ASA Sectional Committee N7. *Radiation Protection.* Sponsors: Atomic Industrial Forum, Inc., 3 East 54th Street, New York 22, N. Y., and National Safety Council, 425 North Michigan Avenue. Chicago 11, Ill. This committee has 22 organization members. It represents ASA on ISO Technical Committee 85-SC2. It studies safety standards for the protection of persons employed in facilities associated with the production and utilization of fissionable materials against the normal, routine hazards present in such facilities (mines, mills, refineries, separation plants, fuel-element-fabrication plants, critical assembly facilities, processing and reprocessing plants, working areas around reactors, and transportation and purification of radioactive materials). Subcommittees are as follows:

- N7-1 Uranium Mines and Mills
- N7-2 Uranium and Thorium Refineries
- N7-3 Isotopic Separation Plants
- N7-4 Health Physics of Fuel-element Fabrication
- N7-5 Health Physics for Reactors

*See the section on Site Selection Criteria in *Nuclear Safety*, 1(2): 2 (December 1959).

†See the section on Reactor Containment in this Review.

Ib

American Standards Association Committees on standards closely allied with nuclear safety but not under the jurisdiction of the ASA Nuclear Standards Board.

ASA Sectional Committee Z54. *Industrial Use of X Rays and Radiation.* Sponsor: National Bureau of Standards. Subcommittees are as follows:

- Z54-1 General Provisions: Methods and Materials of Protection
- Z54-2 Health Provisions and Monitoring
- Z54-3 X-ray Protection for Installations up to 2 Million Volts
- Z54-4 Protection for Installations Above 2000 Kv
- Z54-5 Gamma-ray Sources for Industrial Radiography
- Z54-6 Electrical Protection
- Z54-7 X-ray Diffraction, Fluorescence Analysis, and Microradiography
- Z54-8 Sealed Beta-ray Sources
- Z54-9 Contamination Levels for Industrial Materials

ASA Sectional Committee Z62. *Uniform Industrial Hygiene Standards*

ASA Sectional Committee B31. *Code for Pressure Piping.* This committee has an advisory committee on nuclear piping.

II

American Meteorological Society, Committee on Air Pollution. Chairman: G. R. Hilst, Hanford Atomic Products Operation, Richland, Wash. The aim of this committee is to produce standards regarding the meteorological phase of waste disposal to the atmosphere. It has no formal connection with other bodies. Findings are promulgated through the Society's official publications.

III

American Nuclear Society Standards Committee. Chairman: C. R. McCullough, 15201 Rosecroft Road, Rockville, Md. This committee is composed of an eight-member steering committee and 10 projects (subcommittees). The ANS standards committees work very closely with the ASA nuclear standards committees. ANS has representatives on ASA Committees N2, N3, N4, N5, N6, N7, and the ASA Nuclear Standards Board. ANS cosponsors ASA Sectional Committee N6, *Reactor Safety*, together with ASME.

Table I-1 LIST OF ORGANIZATIONS PRODUCING STANDARDS DIRECTLY INVOLVING SAFETY IN THE USE OF NUCLEAR ENERGY (CONTINUED)

ANS Project 1. *Reactor Classification.* Chairman: U. M. Staebler, USAEC, Washington 25, D. C. The aim of this committee is to develop a standard classification of reactors by type and use for reference in future reactor-standards work. Copies of a proposed standard have been distributed for comments.

ANS Project 2. *Reactor Environment.* Chairman: R. O. Brittan, Argonne National Laboratory, Box 299, Lemont, Ill. (See also N6-1 and N6-2.) The aim of this committee is to establish standards with respect to population, meteorology, geology, proximity to roads, railroads, waterways, containment system, and other factors pertinent to the selection of suitable reactor sites. At the request of ASA, this group met in December 1959 to review its proposed standard on site evaluation.

ANS Project 3. *Reactor Operator Qualifications.* Chairman: M. M. Mann, Division of Inspection, USAEC, Washington 25, D. C. (See also N2-3 and N6-6.) The aim of this committee is to establish standards for evaluating the qualifications of persons as reactor operators of various types and degrees of responsibility.

ANS Project 4. *Reactor Operation.* Chairman: M. M. Mann, Division of Inspection, USAEC, Washington, 25, D. C. (See also N5 and N6.) The aim of this committee is to establish standards with respect to local review boards, inspections, monitoring, maintenance and repair, prestart-up checkouts, rules of conduct, organization records, visitors, and other matters which affect the safety of reactor operation.

ANS Project 5. *Methods of Estimating Energy and Fission Products Release.* Chairman: O. H. Greager, Irradiation Processing Department, Hanford Atomic Products Operation, Richland, Wash. (See also N6-7.) The aim of this committee is to establish standards for estimating probable energy and fission products released in reactor accidents.

ANS Project 6. *Reactor Dynamic System Design.* Chairman: M. M. Shapiro, Nuclear Development Corporation of America, 5 New Street, White Plains, N. Y. (See also N6, especially N6-3 and N6-4.) The aim of this committee is to establish standards for the general design of the over-all reactor complex (cooling systems, safety devices, stability in transient regimes, etc.).

ANS Project 7. *Reactor Components.* Chairman: R. G. Chalker, Atomics International, Box 309, Canoga Park, Calif. (See also N3, N6, and N7-5.) The aim of this committee is to establish safety standards for components of reactor systems, such as power supplies, detectors, preamplifiers, relays, recorders, actuators, pumps, pressure vessels, piping, and valves.

ANS Project 8. *Fissionable Material Outside Reactors.* Chairman: A. D. Callihan, Oak Ridge National Laboratory, Box X, Oak Ridge, Tenn. (See also N6-8.) The aim of this committee is to establish realistic standards for the prevention of unintended critical assemblies.

ANS Project 9. *Nuclear Glossary Subcommittee.* Chairman: J. A. Hunter, Martin Company, Baltimore 3, Md. (See also N1 and N2-4.) ANS is responsible for preparing for ASA the reactor physics section of the Nuclear Glossary and is jointly responsible with ASME for the nuclear fusion section.

ANS Project 10. *Units in the Nuclear Industry.* Chairman: V. L. Parsegian, Rensselaer Polytechnic Institute, Troy, N. Y. The aim of this committee is to establish standard units, with definitions, for use in the nuclear energy field.

IV

American Society of Civil Engineers, Committee on Nuclear Structures and Materials. Chairman: Abbott Frank, ANP Department, General Electric Company, Cincinnati 15, Ohio.

ASCE Committee 1. *Administrative Committee on Nuclear Structure and Materials.* Chairman: Abbott Frank, ANP Department, General Electric Co., Cincinnati 15, Ohio.

ASCE Committee 2. *Hot Laboratories.* Chairman: H. M. Glen, Oak Ridge National Laboratory, Box X, Oak Ridge, Tenn.

ASCE Committee 3. *Live Loads Due to Heat and Incidence.* Chairman: Professor J. E. Goldberg, School of Civil Engineering, Purdue University, Lafayette, Ind.

ASCE Committee 4. *Structural Aspects of Nuclear Insurance.* Chairman: G. Morris, Oak Ridge National Laboratory, Box X, Oak Ridge, Tenn.

ASCE Committee 5. *Structural Materials in Reactor Design.* Chairman: S. H. Fistedis, Argonne National Laboratory, Downers Grove, Ill.

Table I-1 LIST OF ORGANIZATIONS PRODUCING STANDARDS DIRECTLY INVOLVING SAFETY IN THE USE OF NUCLEAR ENERGY (CONTINUED)

ASCE Committee 6. *Nuclear Energy.* Chairman: R. N. Bergstrom, Sargent & Lundy, Arlington Heights, Ill.

ASCE Committee 7. *Construction of Nuclear Facilities.* Chairman: W. C. Siler, Jackson and Moreland, Inc., South Lincoln, Mass.

ASCE Committee 8. *Land Use Aspects of Applied Nuclear Energy.* Chairman: J. C. Callahan, Morris Knowles, Inc., Easton, Pa.

ASCE Committee 9. *The Sanitary Aspects of Nuclear Energy.* Chairman: J. G. Terrill, Jr., c/o Abbott Frank, ANP Department, General Electric Company, Cincinnati, Ohio.

V

American Society of Mechanical Engineers, Committee on Reactor Safety Standards. Standards Manager: Frank Phillippbar, ASME, 29 West 39th Street, New York, N. Y. ASME has representatives on ASA Committees N2, N3, N4, N5, N6, N7, and the ASA Nuclear Standards Board. ASME cosponsors ASA Sectional Committee N6, *Reactor Safety*, together with ANS. ASME prepares and distributes many ASME standards of direct and indirect bearing on nuclear safety, in addition to other standards (e.g., ASME Vessel and Boiler Codes).

VI

American Society for Testing Materials, Special Administrative Committee on Nuclear Problems. Chairman: N. L. Mochel, ASTM, 1916 Race Street, Philadelphia, Pa. The following subcommittees are working on standards in the nuclear energy field:

ASTM Committee A-5. Corrosion of Iron and Steel

ASTM Committee A-10. Iron-Chromium, Iron-Chromium-Nickel, and Related Alloys (six groups)

ASTM Committee B-2. Nonferrous Metals and Alloys

ASTM Committee B-7. Light Metals and Alloys Cast and Wrought

ASTM Committee C-9. Concrete and Concrete Aggregates (with special subcommittee on shielding)

ASTM Committee C-21. Ceramic Whiteware and Related Products

ASTM Committee D-2. Petroleum Products and Lubricants

ASTM Committee D-9 and D-20. Joint Subcommittee on Effects of Nuclear and High-energy Radiation

ASTM Committee D-19. Industrial Water

ASTM Committee E-2. Emission Spectroscopy

ASTM Committee E-3. Chemical Analysis of Metals

ASTM Committee E-10. Radioisotopes and Radiation Effects

ASTM Committee E-14. Mass Spectroscopy

VII

Atomic Industrial Forum, Committee on Codes and Standards. Chairman: Robert Wells. Secretary: G. Edwin Brown, Jr., Atomic Industrial Forum, Inc., 3 East 54th Street, New York, N. Y. Nineteen different segments of industry are represented on this committee. The primary objective of the committee is "to provide leadership on behalf of the Forum in the American Standards Association's nuclear standards programs and to review and comment on proposed federal, state, and local legislation affecting safety and other standard areas in nuclear energy." Summaries of the meetings of the committee are available from the Atomic Industrial Forum, Inc.

The Atomic Industrial Forum is a member of ASA and is represented on the ASA Board of Directors, ASA Standards Council, and ASA Nuclear Standards Board. It sponsors ASA Sectional Committee N2 and cosponsors, with the National Safety Council, ASA Sectional Committee N7. It appointed the U. S. delegates to ISO Technical Committee 85-SC1 and SC2, the international counterparts of ASA N2 and N7.

VIII

Institute of Radio Engineers. IRE has representatives on ASA Committees N3 and N6 and the ASA Nuclear Standards Board. IRE sponsors ASA Sectional Committee N3, which also represents ASA on IEC Technical Committee 45, Electrical Instrumentation.

Committee 13 on Nuclear Techniques. Chairman: R. W. Johnston, Chief, Radiation Instruments Branch, USAEC, Washington 25, D. C. The scope of the standards work affecting nuclear safety includes (1) the selection of terms and the preparation and maintenance of standard definitions in the fields of nuclear radiation and nuclear instrumentation, (2) the preparation and maintenance of standards covering methods of measurement in these fields, (3) the compilation of information

Table I-1 LIST OF ORGANIZATIONS PRODUCING STANDARDS DIRECTLY INVOLVING SAFETY IN THE USE OF NUCLEAR ENERGY (CONTINUED)

relating to operating standards in these fields, and (4) coordination with activities of other IRE committees and other professional societies and liaison with technical organizations engaged in allied work.

IX

National Electric Manufacturers Association, 155 East 44th Street, New York, N. Y., J. F. Miller, Managing Director. NEMA has representatives on ASA Sectional Committees N3, N4, and N6 and on the ASA Nuclear Standards Board. NEMA has developed standards for canned motor pumps for nuclear applications.

X

U. S. Department of Commerce, National Bureau of Standards. A. V. Astin, Director. Publications available from the Superintendent of Documents, U. S. Government Printing Office, Washington 25, D. C. NBS prepares handbooks summarizing the recommendations of the National Committee on Radiation Protection and Measurements (NCRP), which also works closely with the International Commission on Radiological Protection (ICRP). The NCRP was organized and continues to operate under sponsorship of the NBS. Chairman of the main committee of NCRP is L. S. Taylor, National Bureau of Standards, Washington 25, D. C. Forty-one main committee members and 18 subcommittees are listed, with their respective chairmen, in NBS Handbook 69, *Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure,* issued June 5, 1959.

The following NBS handbooks that relate directly to safety in utilization of nuclear energy are available:

42 *Safe Handling of Radioactive Isotopes*
48 *Control and Removal of Radioactive Contamination in Laboratories*
49 *Recommendations for Waste Disposal of P^{32} and I^{131} for Medical Users*
50 *X-ray Protection Design*
51 *Radiological Monitoring Methods and Instruments*
53 *Recommendations for the Disposal of C^{14} Wastes*
54 *Protection Against Radiation from Radium, Co^{60}, and Cs^{137}*
55 *Protection Against Betatron-Synchrotron Radiations up to 100 Million Electron Volts*
57 *Photographic Dosimetry of X and Gamma Rays*
58 *Radioactive-waste Disposal in the Ocean*
59 *Permissible Dose from External Sources of Ionizing Radiation*
60 *X-ray Protection*
61 *Regulation of Radiation Exposure by Legislative Means*
62 *Report of the International Commission on Radiological Units and Measurements (ICRU) 1956*
63 *Protection Against Neutron Radiation up to 30 Million Electron Volts*
64 *Design of Free-air Ionization Chambers*
65 *Safe Handling of Bodies Containing Radioactive Isotopes*
66 *Safe Design and Use of Industrial Beta-ray Sources*
69 *Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure*

XI

Society of Naval Architects and Marine Engineers, Ships Machinery Committee, M-13 Panel. Chairman: Arthur R. Gatewood, American Bureau of Shipping, 45 Broad Street, New York, N. Y. The M-13 Panel prepared Technical Research Bulletin 3-6, "Safety Considerations Affecting the Design and Installation of Water-cooled and Water-moderated Reactors on Merchant Ships," at the request of the U. S. Coast Guard.

XII

United States Committee for the Revision of the International Convention for the Safety of Life at Sea, 1948, Nuclear Power Committee. Chairman: A. R. Gatewood, American Bureau of Shipping, 45 Broad Street, New York, N. Y. A report has been prepared on request of the U. S. Coast Guard for transmission to the U. S. State Department, who presented it to the International Marine Consultative Organization (IMCO) for distribution to the governments who will participate in the 1960 Safety of Life at Sea Conference in London, England.

XIII

Electronic Industries Association, 1721 DeSales Street NW, Washington 6, D. C., J. D. Secrest, Executive Vice President.

Committee TR-19. Committee on Nuclear Instrumentation, Chairman: N. Anton, 1721 DeSales Street NW, Washington 6, D. C.
Committee TR-19.1. Subcommittee on Dosimeters and Chargers
Committee TR-19.4. Subcommittee on Film Badge Dosimetry

Table I-2 LIST OF ORGANIZATIONS ACTIVELY WORKING IN THE PRODUCTION OF STANDARDS WITHIN THE AMERICAN STANDARDS ASSOCIATION

The following is a list of 62 organizations actively working in the production of standards within the American Standards Association. Appearing in parentheses alongside the organization name are numbers indicating the committee or committees of which the organization is a member.

Aircraft Industries Association*†
American Bureau of Shipping (N6)
American Chemical Society† (N2,N3,N5,N6,N7)
American Conference of Governmental Industrial Hygienists (N3,N5,N6)
American Federation of Labor and Congress of Industrial Organizations† (N6,N7)
American Hospital Association (N6)
American Industrial Hygiene Association† (N2,N5,N6, N7)
American Institute of Chemical Engineers† (N3,N5,‡ N6)
American Institute of Electrical Engineers† (N3,N4,‡ N6)
American Institute of Mining, Metallurgical and Petroleum Engineers (N6)
American Institute of Physics (N6)
American Iron and Steel Institute (N6)
American Medical Association (N6)
American Municipal Association (N2)
American Nuclear Society† (N2,N3,N4,N5,N6,‡ N7)
American Public Health Association† (N5,N6,N7)
American Society of Civil Engineers† (N5,N6)
American Society of Mechanical Engineers† (N2,N3, N4,N5,N6,‡ N7)
American Society of Safety Engineers† (N7)
American Society for Testing Materials† (N3,N5,N6)
Associated General Contractors of America (N6)
Association of Casualty and Surety Companies† (N2, N5,N6,N7)
Association of State and Territorial Health Officers (N2,N7)
Atomic Industrial Forum† (N2,‡ N3,N6,N7‡)
Bureau of Explosives*†
Compressed Gas Association (N6)
Conference of State and Provincial Health Authorities*†
Electric Light and Power Group (Edison Electric Institute and Association of Edison Illuminating Companies)† (N4,‡ N5,N6)
Electronic Industries Association† (N3,N4)
Elliot Company, Division of Carrier Corporation (N4)
General Electric Company (N4)
Health Physics Society† (N2,N3,N5,N6,N7)
Institute of Radio Engineers (N3,‡ N6)
Instrument Society of America (N3,N6)
International Association of Government Labor Officials† (N2,N7)
International Association of Machinists (N6)
International Brotherhood of Electrical Workers (N3, N4,N6,N7)
Mallinckrodt Chemical Works (N5)
Manufacturers Standardization Society of the Valves and Fittings Industry† (N6)
Manufacturing Chemists' Association† (N3,N5,N6,N7)
Mechanical Contractors Association of America (N6)
National Association of Corrosion Engineers (N5)
National Association of Mutual Casualty Companies† (N2,N5,N6,N7)
National Association of Plumbing Contractors (N6)
National Board of Fire Underwriters (N5,N6)
National Bureau of Standards† (N2,N3,N4,N6,N7)
National Clay Pipe Manufacturers Association (N6)
National Electric Manufacturers Association† (N3, N4,‡ N6)
National Industrial Zoning Committee (N6)
National Research Council (N6)
National Safety Council† (N2,N7‡)
Scientific Apparatus Makers Association† (N3,N5,N6)
Society of Automotive Engineers† (N3,N5,N6,N7)
Society of Naval Architects and Marine Engineers (N6)
Underwriters Laboratories† (N3,N4,N5,N6,N7)
U. S. Atomic Energy Commission† (N2,N5,N6,N7)
U. S. Department of the Army (N4)
U. S. Department of the Army, Corps of Engineers (N3)
U. S. Department of Defense† (N2,N5)
U. S. Department of Health, Education, and Welfare, Public Health Service† (N2,N5,N6,N7)
U. S. Department of Labor, Bureau of Labor Standards, Industrial Safety Standards Division† (N7)
U. S. Department of the Navy, Bureau of Ships (N3)

*This organization is not represented on any individual ASA Committees.

†This organization is a member of, or is represented on, the ASA Nuclear Standards Board.

‡This organization is the sponsor (or cosponsor) of the indicated ASA Committee.

ferent organizations. However, the same general method is followed by almost all producers of published standards. A committee is directed to prepare a proposed standard. After a study of the proposed standard, it is submitted to the cognizant body of the organization and to other bodies or committees having related interests for comment and criticism. The originating committee then prepares a final draft for approval of the cognizant body. When the standard is approved, it is published and made available to the public. Frequently the standard is then presented to ASA for consideration as an American standard. Approved American standards are often presented by ASA to international bodies (e.g., ISO, IEC, and PASC) for consideration as international standards.

In instances where the report of a particular standards committee is available, this fact is indicated in Table I-1. In all other instances, it may be presumed that a committee report has not been issued. Information on the status and scope of committee endeavors may be obtained by contacting the responsible chairman.

(L. A. Mann)

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Reactor Accidents

Of the several fields of investigation which are directly related to nuclear safety, perhaps the most important one deals with the kinetics of reactors. The mechanisms for investigating the kinetics problems, both in theory and experiment, must be highly developed and conscientiously applied for each reactor system before the first full-scale power plant of that type is operated. The kinetics problems of water-moderated reactors (especially light water) are emphasized in this review.

Light-water-moderated Reactors

The fourth in a series of experiments (BORAX-IV) to examine the characteristics of strongly heterogeneous systems designed to simulate boiling-water power reactors has been described by Maxon et al.[1] The fuel assembly was intended to mock up a possible design for use in a central power station. In the mock-up the fuel was in the form of enriched urania-thoria pellets bonded with cast lead inside extruded aluminum-tube plates (tubes extruded together to form platelike shapes). There was a total of 47 fuel-containing tubes in each assembly, as well as one tube containing boron steel. Each assembly measured 3.87 by 3.83 in. and had an effective (uranium-containing) length of 24 in. The number of assemblies in the core varied from 28 at atmospheric conditions to 59 at 420°F and 300 psig. Four shim and shutdown rods provided control, and a fifth very small rod provided reactivity changes for transients. The flux distribution, power calibrations, and reactivity effects under static conditions were determined for the assembly in the conventional manner.

The experimental studies of major interest to nuclear safety were those concerned with the conditions affecting stability of the system. An attempt was made to produce a means of investigating system transient behavior without high-power surges. There were two conditions which produced undamped oscillations that required the use of shutdown rods to terminate the experiment. In one instance, 0.5 cycle/sec oscillations were produced by a boiloff of the water head above the fuel element, a situation which could be encountered following a sudden loss in pressure such as would be produced by a steam leak. This unstable condition raises a question as to what would be the shutdown mechanism upon failure of the safety-rod system. The effect of water-head boiloff at higher pressures was not covered in the report.[1]

The second condition, which was encountered at essentially all pressures, is described as a power threshold for self-induced oscillations. Here the resonant frequencies of oscillations increased with power above the threshold and, in one case, produced a period as short as 0.5 sec for a sufficient time to produce a period scram.

Additional experiments were performed with a rod oscillator to evaluate the reactor transfer function with respect to power level. A sharp resonance was found in this transfer function near the self-induced oscillatory frequency. This finding will be of value in establishing a technique for future development work and for central-power-station prepower safety evaluation.

Reference 1 contains much information related to the particular reactor design which is used to indicate the probable cause of the oscillations; however, this indication of a mechanism is not yet sufficiently strong to assist in designing for increased safety. (Dunlap Scott)

The self-limiting characteristics of prompt-critical light-water-moderated and -cooled reactors are associated with the liberation of energy in the water. The original Fuchs model[2] for bursts assumed the reactivity compensation to be proportional to the energy produced. However, the model gives only a qualitative fit to the experimental burst shape. Forbes et al.[3] have proposed a model in which the reactivity com-

pensation at time t is proportional to the energy produced up to some previous time $(t - \tau)$ raised to the nth power, as shown in Eq. 1:

$$\frac{1}{\phi(t)}\frac{d\phi}{dt} = \alpha_0 - bE^n(t-\tau) \qquad (1)$$

where $\phi(t)$ is proportional to reactor power at time t, α_0 is the inverse of the initial reactor period, b is the energy coefficient of reactivity, $E(t-\tau)$ is the energy liberated up to time $(t-\tau)$, and n and τ are parameters which are adjusted to fit the shape of the burst. This model has been successfully used to fit the shape of prompt bursts in the SPERT reactor. The best fit has been obtained with values of n between 1.5 and 2.5 and values of τ much greater than $1/\alpha_0$.

Based on Eq. 1, the maximum power during a burst is given by

$$\phi_{\max} = \frac{\alpha[(n+1)/n]}{b(1/n)}\left[e^{\alpha\tau} - (1/n)\right] \qquad (2)$$

If n were taken as 2, the peak power would vary as $b^{-1/2}$. This type of behavior was demonstrated in the SPERT experiments for several core configurations.

The maximum power level reached as a result of a ramp addition of reactivity can be related to an equivalent step addition of reactivity. If the minimum period exhibited during a ramp addition is equal to the initial period of a step addition of reactivity, the peak power levels are approximately equal.

The strong self-limiting characteristics of light-water-moderated and -cooled reactors protect them from injurious start-up accidents. Investigations of start-up accidents in the Yankee reactor[4] have shown that the reactor will not be damaged during the initial power burst. Following this initial burst there is a relatively slow rise in reactor power, and the slope of this rise has been shown to depend on the magnitude of the Doppler and moderator coefficients of reactivity. If corrective action is taken within 25 sec of the initial burst, no serious damage will be done to the reactor. (S. Jaye)

In a paper by Ledinegg,[5] the dynamic equations of a boiling-water reactor are set up under a number of simplifying assumptions, not all of which are stated explicitly. For instance, it is assumed that steam generation is constant over the reactor, that the transit time of the water through the reactor is short compared with the reactor oscillation, and that the heat transfer from the fuel into the water depends linearly on the temperature difference between the fuel and the water. Also, all other equations are linearized. The negative fuel temperature coefficient resulting from the Doppler effect is completely neglected, although this is an important effect in many reactors of the boiling-water type. There are also a few minor mistakes in the equations; for example, the capture cross section is used where the fission cross section is meant. The main conclusions are, however, in accordance with generally known facts: (1) boiling-water reactors may be unstable; (2) instability is increased by decreasing the operating pressure and increasing the transit time of the water, i.e., decreasing its flow rate; and (3) boiling-water reactors, with both natural and forced circulation, can be designed to be stable.

Data obtained from BORAX-I to -IV, EBWR, LITR, VBWR, and SPERT-I experiments are discussed in an Argonne National Laboratory report.[6] In particular, the following characteristics of the dynamic behavior of these reactors are examined: power, pressure, temperature, reactor amplitude, rate of change of these variables, thresholds, and harmonic content. It is pointed out that these reactors have a continuous transition from extremely stable to extremely unstable conditions, unlike electronic oscillators, which either oscillate or fail to oscillate.

A brief discussion of the theory of the transfer function is followed by short discussions of the following methods of measuring the functions: control-rod oscillator, ringing, and autocorrelation. A critical evaluation of these methods is given, and the discussion is made extremely valuable by the judicious presentation of experimental data.

The kinetic theories are discussed next, starting with the simplest, the phenomenological feedback, and proceeding through a simplified physical model to transfer functions calculated by IBM-650 and -704 computers. Hydraulic instability, which is known experimentally, is briefly discussed, and the so-far-hypothetical flux-tilt instability is mentioned.

Appendixes give reactor mechanical data and data on reactor physics for BORAX-II, -III, and -IV, some data for the EBWR, another short discussion of reactor hydraulics, a discussion

of harmonic content in the power variation, and a generalized theory of feedback.

(W. K. Ergen)

Graphite Oxidation

In the operation of gas-cooled reactors moderated by graphite, excessive oxidation of the graphite by the coolant itself, by impurities in the coolant, or as a consequence of a loss-of-coolant accident is recognized as a potential hazard. Operating temperatures are therefore selected at which the rates of oxidation of the graphite are sufficiently low to permit lifetime operation of the reactor without excessive oxidation of the moderator and accompanying losses of moderating capacity and mechanical strength. Damage to the graphite, or even uncontrolled oxidation, may result if the coolant temperatures greatly exceed the design temperatures as a result of reactor excursions or annealing of the graphite. The latter process is of prime concern for reactors operating at low graphite temperatures since the quantity of energy stored in the graphite as a result of prolonged irradiation may, under certain conditions, be sufficient to raise the graphite temperature to a hazardous level when the low-temperature component of the stored energy is released by annealing. The stored energy in the graphite is not considered a serious problem in gas-cooled reactors using CO_2 as the coolant. Not only is the accumulated energy considerably less than that with other coolants, because of the higher graphite temperatures used in these systems, but also, during annealing, there is no possibility of an uncontrolled reaction (excluding a rupture in the coolant system) because the reaction of graphite with CO_2 is endothermic. Prolonged operation at too high coolant temperatures may, however, result in excessive oxidation of the graphite by CO_2, as well as deposition of carbon in the cooler regions of the reactor. This latter effect may be sufficient to affect the heat-transfer rates, or even the flow rates, and thereby produce operational problems. These considerations are important in determining the design life of the graphite core structure. The oxidation that would ensue following a loss-of-coolant accident, on the other hand, would play an important role in determining the resultant temperature excursion and activity release. In this latter event the replacement of a relatively "inert" coolant, such as helium or CO_2, with air might result in the production of significant chemical energy, which, in addition to the fission-product heat and stored heat in the fuel, would have to be removed in order to prevent a runaway situation.

Measurements of Reaction Rates in Air and Oxygen

The reaction of oxygen or air with graphite is strongly exothermic and has a large positive temperature coefficient, i.e., properties that would contribute to an uncontrolled or runaway reaction. Reaction rates have been measured by a number of investigators,[7,8] and the results have differed widely, probably, in large part, because of differences in the properties of the graphites used. Values reported by these investigators for the energy of activation of the reaction C (graphite) + O_2 = CO_2 also vary, but a probable value is about 39 kcal/mole in the absence of radiation.

The fact that radiation promotes the oxidation at relatively low temperatures is well known. The mechanism of the radiation-induced oxidation is complex, and the effects of various parameters are not well understood. As a result, experimental data frequently are difficult to interpret. The rates of oxidation of irradiated graphite and unirradiated (virgin) graphite by oxygen have been measured at the Brookhaven National Laboratory[9] in the absence of radiation, in-pile, and in the presence of a gamma-ray source. Type AGOT graphite that had been exposed to an integrated thermal-neutron flux of 4×10^{20} neutrons/cm^2 (exposure temperature of 25 to 50°C) was found to oxidize in the absence of radiation at a rate of 0.28 per cent weight loss in 100 days at 300°C, compared with a 0.056 per cent weight loss for unirradiated graphite. The ratios of the reaction rates of irradiated graphite to the reaction rates of unirradiated graphite decrease with increasing temperature, reaching a value of about 2.3 at 450°C. Virgin graphite gave weight losses of 0.065 and 0.14 per cent, respectively, in 100 days when oxidized at 300°C in the presence of gamma fluxes of 200,000 and 600,000 r/hr. For comparison, a weight loss of 1.14 per cent in 100 days was measured for previously irradiated graphite oxidized at 300°C in the presence of a gamma flux of 200,000 r/hr. Similar effects were found at 400°C. It is evident that prior ir-

radiation of the graphite increases the reaction rates quite markedly. This effect continues to exist even after 20 to 25 per cent of the graphite specimen has been oxidized. The accelerating effect of gamma rays is less pronounced and probably results from ionization of the gas phase (ozone production) rather than from changes in the properties of the graphite, as is the case for neutron irradiation.

Some interesting results were obtained when unirradiated samples of graphite were oxidized in-pile in the temperature range 250 to 400°C, using a flux of 1×10^{12} or 4×10^{12} neutrons/(cm^2)(sec). The ratio of reaction rates measured at the high flux to those measured at the low flux varied from about 10 at 250°C to less than 2 at 400°C. At 250 and 300°C the rates found for oxidation in the presence of the high flux were higher than those noted for irradiated graphite oxidized in the absence of radiation. At 350 and 400°C, however, the reverse was found. In fact, at 400°C the oxidation rate found in the presence of the high flux was lower than that observed for unirradiated graphite in the absence of radiation by a factor of about 6, as well as being lower than the oxidation rate found at 300°C in the presence of the high flux. Oxidation rates measured in the presence of the low flux were lower at all temperatures than those found for irradiated graphite in the absence of irradiation and also were lower at 350 and 400°C than the rates found for virgin graphite oxidized in the absence of radiation at these temperatures. Activation energies for the reaction in the low flux and the high flux are 9.0 and 1.6 kcal/mole, respectively, compared with 48.8 kcal/mole for the virgin graphite and 36.1 kcal/mole for the irradiated graphite when oxidized in the absence of radiation. These effects are, as the investigators say, "startling" and not understood. No other in-pile studies have covered this temperature range, and therefore it is not known whether these results are confirmable.

Calculated Rates of Oxidation in Reactor Channels

It probably should not be surprising that large differences in the in-pile rates of oxidation of graphite have been reported since the processes involved are complex and influenced by many factors, such as temperature, pressure, flow rate, irradiated gas volume, and reactor flux. In addition, the diverse results obtained in-pile for small specimens suggest that it would be unwise to extrapolate from these results to predict the oxidation rates that would prevail in a reactor core. Tomlinson[10] has made a critical comparison of the data collected from studies in several reactors in an attempt to reconcile some of the different rates of oxidation reported. The data compared came from experimental studies made in the NRX reactor, Windscale Pile 1, and BEPO. Tubular specimens of graphite were oxidized in air at 150 to 300°C in the NRX reactor. Graphite rods spaced along an "isotope" channel were oxidized by the reactor cooling air flowing over them at temperatures varying from 20°C at the inlet to 160°C at the outlet of the channel in Windscale Pile 1. Oxidation of graphite rods and tubes in BEPO was carried out in a closed silica system using oxygen. The rate of CO_2 production was measured for determining oxidation rates in the BEPO experiment, whereas weight losses of specimens were used in the other studies.

Oxidation rates in the range 0.024 to 0.064 per cent weight loss following a thermal-neutron dose to the graphite of 10^{20} neutrons/cm^2 were found in the Windscale experiments; the specimens were irradiated at temperatures of 70 to 100°C. For the NRX experiments the values ranged from 0.29 to 0.45 per cent weight loss for irradiation to the same dose at temperatures from 150 to 230°C; and in the BEPO experiments the oxidation rates ranged from 1 to 4.5 per cent weight loss, also for irradiation to the same dose at temperatures from 60 to 250°C. At 200°C the rate of oxidation in the absence of radiation is about 5×10^{-4} per cent weight loss in 1000 days, which is the time required for a dose of 10^{20} neutrons/cm^2 in a flux of about 10^{12} neutrons/(cm^2)(sec). Small variations in the radiation intensities would not account for the large differences in oxidation rates, and consequently the flow rates and geometry factors must be responsible. The flow rates were about 40 liters/sec in the Windscale experiments, 14 liters/sec in NRX, and only 0.017 liter/sec in BEPO.

A prior report[11] on the BEPO experiments stated that, at the temperatures of interest, the graphite was oxidized by ozone formed in the irradiated gas and that this process was more important than the radiation effects on the graphite itself. The over-all reaction rate measured was the sum of the oxidation rates of the exter-

nal graphite surface and of the internal, or pore, surface. The rate of oxidation of the external surface is believed to be strongly dependent on the flow rate and the volume of gas irradiated upstream of the graphite, whereas the reaction rate in the pore structure is independent of these parameters. Tomlinson[10] calculated the graphite weight losses at 200°C (for a total dose of 10^{20} neutrons/cm^2 in the graphite) from surface oxidation by the ozone formed by radiolysis of the gas upstream of the graphite, using the yield value for the production of ozone, the reversion rate of ozone to oxygen, the fraction of ozone reacting with the graphite, and the yield of the ozone-graphite reaction.[9] The calculations were made for the parameters that prevailed during the experimental studies in the three reactors, and it was assumed that the over-all rate of oxidation was not limited by the diffusion of ozone from the gas to the graphite surface. Also, the rate of oxidation by the ozone produced in the pore structure was calculated using yield values for the production of ozone and for the ozone-graphite reaction;[9] it was assumed that the removal of CO_2 was rapid enough to prevent the build-up of any significant concentration. The oxidation rate in the pore structure, which varies only with the temperature and total radiation dose, was found to be 0.036 per cent weight loss for a dose of 10^{20} neutrons/cm^2 to the graphite at 200°C. The temperature dependence of the NRX results was used to extrapolate the measured oxidation rates of the Windscale and BEPO studies to 200°C for comparison with the calculated values. Agreement between the calculated and experimental values to within a factor of 4 or less was obtained.

The effects of the flow rates on the oxidation rates of the graphite samples under the conditions present in the three reactors may be summarized as follows:

1. The high-velocity air used in the Windscale experiments removed most of the ozone before reaction occurred at the graphite surface, and therefore the oxidation was primarily in the pore structure.

2. In the BEPO experiments, the very low flow rate resulted predominantly in surface oxidation, even though considerable ozone was lost by reversion to oxygen before it had an opportunity to react with the graphite.

3. The intermediate flow rate used in the NRX experiments resulted in a surface oxidation rate lower than that found in the BEPO experiments and higher than that of the Windscale experiments. More surface oxidation occurred, however, than pore structure oxidation.

The marked effects of flow rates and the geometry factors on the oxidation rates of graphite specimens in the presence of reactor radiation point out some of the difficulties encountered when the experimentally determined oxidation rates are used to predict the losses of graphite in the moderator structure of an air-cooled reactor.

In an air-cooled reactor an increase in the graphite temperature (because of an increased air inlet temperature, decreased coolant flow, or release of stored energy) could result in uncontrolled thermally induced oxidation of the graphite and extremely high graphite and air temperatures. To prevent the occurrence of such an event, some means must be provided to remove the large quantity of heat generated by the strongly exothermic and temperature-sensitive reaction of air with graphite. The air flow through a long graphite channel favors a runaway since a rise in air temperature normally results as it flows through the channel. British investigators[12] have made both theoretical and experimental studies of the conditions leading to runaway oxidation of graphite by air flowing through a graphite channel. As a model, the theoretical treatment used a regular graphite lattice with parallel channels through which air flowed. Longitudinal heat transfer within the graphite and temperature gradients perpendicular to the channels were ignored. The mathematical analysis permits (1) calculations of the length of channel required to give critical conditions (air and graphite temperatures above which instability results) for given inlet conditions, (2) calculations of the inlet graphite and air temperatures which produce instability at the outlet end for a specified channel length, and (3) calculation of the distribution of graphite and air temperatures along the length of the channel for given inlet conditions. In addition, experimental studies were performed to verify the predicted inlet temperatures for specified runaway conditions and the equilibrium temperature distribution along a channel. The "graphite burning rig" used contained a vertical stack of eight blocks of graphite having a series of $\frac{1}{4}$-in. holes on a 1-in.-square lattice. Guard heaters were used for minimizing lateral heat transfer, as well as for the initial heatup. Preheated air

was passed up the stack at the desired inlet temperatures and flow rates. It was hoped that a stable temperature distribution could be achieved. This proved to be impossible, however, because of lateral heat losses. A decrease in the temperature at all points was noted for an air inlet temperature of 520°C at a flow rate of 60 liters/min (measured at 25°C), whereas an increase occurred at an inlet temperature of 530°C with this flow rate. With a flow rate of 170 liters/min, decreases in temperature were observed for air inlet temperatures of 520 and 530°C, but an increase was noted at an air inlet temperature of 540°C. Thus the minimum temperatures required to produce a runaway lie within these temperature ranges given for the two different flow rates, and they are relatively insensitive to changes in flow rates. The theory predicts minimum air inlet temperatures of 475 and 510°C for flow rates of 60 and 170 liters/min, respectively. The higher temperatures found experimentally are believed to be due to lateral heat losses. It was noted that an increase in the rate of air flow is not effective in stopping a small runaway unless the air inlet temperature is decreased. Also, after stopping such a runaway by lowering the inlet temperature, the temperatures of the upper blocks were still increasing 2 hr later. These observations indicate that control of a runaway by means of the air inlet flow and temperature is difficult and that, even after control is established, damage may continue to occur in parts of the channel. In the last recorded run, a runaway was allowed to continue for several hours. Very rapid increases in temperatures were noted above 700°C, with the most rapid rises occurring in the upper blocks. The thermocouples failed at 1185°C, but it is believed that considerably higher temperatures were reached because the graphite burned with a "large blue flame." These observations point out that, once a large runaway is in progress, air flow probably will be ineffective in halting it. Some drastic measure, such as the introduction of steam or CO_2, would seem to be required to control the runaway. It should be noted that the results cited were obtained using unirradiated graphite, and it is probable that lower burning temperatures would prevail for irradiated graphite.

Measurements in Carbon Dioxide

The reaction C (graphite) + CO_2 = 2CO has a large positive-temperature coefficient, but, since it is endothermic, no uncontrolled reaction is possible. Reaction rates have been reported by a number of investigators[13–15] for the gasification of graphite with CO_2. The rates obtained in the absence of radiation vary considerably, but they are lower than the oxidation rates obtained for air. For comparable rates of oxidation, the CO_2 reaction must be carried out at temperatures several hundred degrees higher than those required for the air reaction. Radiation has an accelerating effect on the reaction rate,[16–18] which is probably determinable up to about 500°C. The thermally induced reaction would be expected to mask any radiation-induced effects at higher temperatures. It is believed that the initial step in the radiation-induced reaction involves decomposition of CO_2 to yield ozone and that, after this initial step, the reactions are similar to those occurring in the radiation-induced air-graphite reaction. The radiation-induced CO_2-graphite reaction has a very small positive-temperature coefficient, and, like the air-graphite system, the reaction rate is sensitive to flow rate and the volume of gas irradiated upstream of the graphite.

British investigations[18] performed in the Pressurized Pile Producing Power and Plutonium (PIPPA) model channel in BEPO and in the Calder Hall reactors have produced some interesting information on this radiation-induced reaction. The model channel experiments were run to provide information relative to the conditions prevailing in the Calder Hall reactors. Subsequent operation of Calder Hall reactors 1 and 2 has yielded data on the rate of build-up of CO, as well as the steady-state concentration. Test specimens of graphite removed from the reactor after operation for various periods of time have given rates of weight losses at various positions within the core. While Calder Hall reactor 2 was operating with an outlet gas temperature of 336°C and a mean thermal-neutron flux of 5×10^{12} neutrons/(cm^2)(sec), a steady-state concentration of CO of about 0.4 vol.% was measured in the coolant. The steady-state concentration of CO in Calder Hall reactor 1 is somewhat higher because of a lower leakage rate. A mean rate of weight loss of 0.07 wt.% per year for the moderator in Calder Hall reactor 2 was calculated from the concentration of CO in the coolant and the leakage rate. This compares favorably with a value of 0.09 wt.% per year loss calculated from the weight losses given by various test specimens of graphite

located within the core. The latter corresponds to a loss of about 0.5 ton of graphite per year from the core. These values are in good agreement with the values predicted by data from the PIPPA model channel studies. No positive evidence of carbon deposition has been found in any region of the Calder Hall reactors.

British experience indicates that, under the operating conditions of the Calder Hall reactors, the reaction of CO_2 with graphite does not result in either excessive burnout of graphite or carbon deposition. The anticipated operation of gas-cooled reactors, with helium as the coolant and with graphite at temperatures in the 600 to 1000°C range (i.e., a range in which the rates of oxidation of graphite by CO_2 are many times higher than those in the Calder Hall reactors), poses stringent limitations on the concentration of CO_2 present in the coolant if excessive damage to the graphite and carbon deposition are to be avoided. In addition, the high graphite temperatures can produce hazardous effects in the event of air leakage into the coolant system.

(L. G. Overholser)

Fission-product Escape from Uranium-Graphite Fuel Materials

The potential hazard associated with a nuclear reactor is directly dependent upon the degree to which the disposition of the fission products can be controlled during normal operation and at the time of an accident. In those reactors which have been approved for construction and operation, the fission-product activity is generally confined within the fuel elements during normal operation since the fuel bodies retain fission products, and any activity which does escape from the fuel body is contained by an impervious metal capsule. (A notable exception is, of course, the fluid-fuel type of reactor.) Reactors that use metal-clad fuel elements are limited in performance by the properties of the cladding materials. In order to attain higher temperatures, which tend to result in substantial reductions in power costs, designers of nuclear reactors have proposed fuel elements that contain only ceramic materials, including graphite. Although the mechanical properties of the ceramic materials permit operation at high surface temperatures, the high temperatures and the porosity of the materials introduce the problem of fission-product escape into the primary cooling system. The presence of fission-product activity in the coolant not only increases the hazard associated with leakage of coolant or a rupture of the reactor but also introduces additional hazards associated with performing maintenance on contaminated equipment.

The attractiveness of the potential economic advantage offered by high-temperature systems has led to several studies and reactor proposals in the United States, including a plan for a small reactor experiment, as well as work in Great Britain and elsewhere abroad. Most of the interest at present is in graphite-uranium fuel elements, with helium as the coolant. The discussion which follows is concerned, in general, with systems of that type and, specifically, with fission-product escape from uranium-graphite fuel.

Three recent Oak Ridge National Laboratory (ORNL) reports[19–21] have discussed the release of fission products from graphite-uranium fuel materials. The first report[19] describes some preliminary work which is presented in greater detail in the second report, a design study of the Hot Gas Cooled Reactor No. 1 (HGCR-1).[20] Essentially the same data on fission-gas release from uranium-graphite fuel material are used in the third report,[21] in which some of the problems related to contaminated systems are discussed in greater detail. An attempt was made in these ORNL studies to provide a quantitative basis for evaluating the problem of activity escape from the fuel material into the coolant. The assumptions made for these studies, as well as the results, are indicated later in this Review.

For several years the Sanderson & Porter Company has been investigating the feasibility of reactors in which the core is a bed of fuel-bearing graphite spheres.[22] As a part of this "pebble-bed" reactor program, a design study of a small experimental reactor has been undertaken. The experimental reactor would furnish information on the extent of fission-product leakage from the fuel elements, the feasibility of continuous removal of fission products from the gas stream, system decontamination, and other useful data.[23] As a part of the development program for the pebble-bed reactor, Battelle Memorial Institute is investigating fission-gas release from various types of graphite fuel elements,[23] including spheres formed from an admixture of small UO_2 particles in graphite, lumps of admixture material coated with graph-

ite, and uranium-impregnated graphite spheres. Coatings of Si-SiC and SiC are being studied.

The Los Alamos Scientific Laboratory expects to begin construction shortly of the 3-Mw TURRET reactor, which will use unclad fuel elements of uranium-impregnated graphite.[24] Since the helium coolant is expected to reach 2400°F, TURRET will provide direct experience on fission-product release and deposition and on decontamination techniques.

Release of Fission Products in the HGCR-1

The fissioning of uranium results in the formation of about 300 nuclides totaling 37 elements. The amount of a particular species which exists in a reactor is related to the half lives and fission yields of the nuclide and its precursors. The amount of the nuclide in the coolant depends on the ability of it and its precursors to escape from the fuel body, as well as on the rates of their removal from the coolant.

The discussion of fission-product release in the HGCR-1 report[20] was limited essentially to gamma emitters of energy greater than 0.5 Mev. Some of the other factors taken into account in restricting the number of nuclides considered were (1) mass yield, (2) half life, and (3) diffusion rate. The complexity required in an analysis of fission-product release depends, of course, on the problem being studied. For example, an analysis of shielding requirements requires a study of only the gamma emitters, whereas an analysis of leakage, maintenance, and over-all hazards should include both beta- and gamma-emitting nuclides. The HCGR-1 study does not treat the problem of the spread of activity following system rupture or during the removal or maintenance of reactor components, although this is perhaps one of the most important considerations in a contaminated reactor.

Three basic types of fuel element were considered in the studies of fission-product escape: (1) bulk UO_2 coated with graphite, (2) UO_2 particles dispersed in a graphite matrix, and (3) graphite impregnated with uranium. The technique used for estimating fission-product escape was similar to that developed at Westinghouse[25,26] and at Chalk River[27–29] for fission-gas release from UO_2. In the first type of element, the UO_2 is considered to consist of a mass of dense particles connected by an open-pore structure. Fission products diffuse from the interior of the solid and are evolved into the pores, where they escape from the fuel body by gas-phase diffusion.[26,27] The pores in the UO_2 and graphite offer little resistance to diffusion relative to that involved in escape from the solid; hence the gas release is controlled by the rate of evolution from the solid.

In the initial ORNL studies,[20] the small UO_2 particles in the admixture elements were treated as being porous UO_2 spheres; but, in the later studies,[21] it was pointed out that the particles should be treated as a nonporous structure. In addition to escape by diffusion, some of the fission fragments that originate near the particle surface recoil out of the UO_2 and into the surrounding graphite. Those which terminate within the graphite grains must then diffuse through the solid graphite to the pores in the graphite structure.

The fission fragments formed in impregnated elements are distributed through the structure by recoil. Experiments at North American Aviation[30] have indicated that about 35 per cent of the fission fragments in uranium-impregnated graphite can be leached from the element and that these fragments are, presumably, within the uranium carbide on the pore surface or within the pores themselves. Those fragments which cannot be extracted are supposedly within the graphite grains. The fission products that are in the solid graphite escape by solid-state diffusion, as described for the UO_2. The behavior of the 35 per cent that can be leached is discussed later in this Review.

At the time the HGCR-1 study was made, little information on diffusion rates of fission products in UO_2 existed, except that for the noble gases. In the HGCR-1 analysis, therefore, the data reported by Westinghouse[26] were used for xenon and krypton; and the diffusion coefficients for bromine, iodine, cesium, and rubidium (which also had been found to escape from UO_2) were, for lack of a better procedure, taken to be the same as the diffusion coefficient for krypton. The data used to estimate the diffusion parameters for graphite were obtained from the experiments at North American Aviation, in which the escape of fission products from impregnated graphite at high temperatures (1500°C and above) was investigated.[30–33] In analyses of the admixture and impregnated elements, the HGCR-1 study included the diffusion of cesium, bromine, iodine, strontium, xenon, barium, tellurium, and yttrium through graphite. In ex-

trapolating the diffusion coefficients to lower temperatures, the activation energy was assumed to remain constant, and, in addition, the graphite in the fuel element was assumed to have similar properties to those of the graphite used in the North American Aviation experiments. The uncertainties associated with the diffusion parameters used in the HGCR-1 studies should be recognized since the results could be appreciably in error.

The fuel elements selected for the HGCR-1 design study consisted of 200-μ-diameter spherical UO_2 particles dispersed in graphite plates coated with UO_2-free graphite. The hard-gamma activity released from these elements to the coolant system after long-term operation without processing was estimated to be about 9.7×10^6 curies, i.e., less than 0.3 per cent of that in the reactor. However, the percentage of each activity which would escape varies widely among the nuclides and ranges up to 11.3 per cent for Ba^{137m}. Other long-lived, but not hard, gamma activities estimated to be in the coolant system were 3.0×10^5 curies of 10.3-year Kr^{85}, 6.5×10^5 curies of 28-year Sr^{90}, and 17.2×10^5 curies of 26.6-year Cs^{137}.

For comparison with the activity released from fuel elements containing UO_2 particles in graphite, a tabulation was made of the release of fission products from uranium-impregnated graphite and from UO_2 clad with graphite. Table II-1 summarizes the results of references 20 and 21, except that the value for the impregnated fuel has been adjusted by the reviewer as described below.

Table II-1 COMPARISON OF HARD-GAMMA ACTIVITY RELEASED FROM VARIOUS TYPES OF FUEL WITH THE SAME TEMPERATURE STRUCTURE

Method of fuel-element fabrication	Activity in reactor coolant system, %
Admixture	0.3
Impregnation	3.0
UO_2 clad with graphite	0.1

In comparing the different fuels, it should be noted that the same temperatures were assumed for all. Graphite-clad UO_2 elements would probably have much higher average temperatures than the others for the same surface temperature, and the gas release could be much greater.

The results for graphite-clad UO_2 were obtained by assuming that the effective UO_2 particle radius was the same as that used in reference 20 in estimating the diffusion from UO_2 particles in the admixture. Thus the activity is the same as that which diffuses from UO_2 in the admixture case. As noted previously, the particle radius, as well as the diffusion coefficient, D, was different for the admixture fuel material in reference 21. The activity escape from the impregnated graphite was taken as 25 times that which escapes from the admixture by recoil and subsequent diffusion through the graphite since the fission fragments that escape from 200-μ particles by recoil represent, on the average, only $\frac{1}{25}$ of the total fission fragments. It is possible that the assumption that all the fission fragments enter the graphite is much too optimistic. The North American Aviation studies indicated that about 35 per cent of the fission products could be leached from impregnated fuel and that, when exposed fuel was annealed, between 20 and 40 per cent of some fission products were immediately released.[30–33] If an appreciable fraction of the fission products are near the surfaces of pores, then the activity that would escape is much higher than the 3 per cent given in Table II-1.

Data obtained at Battelle Memorial Institute[22] on impregnated and admixture elements indicate higher gas release rates from the impregnated elements by an order of magnitude, which agrees with the comparison given in Table II-1. However, the experiments also provided evidence of high initial release rates from impregnated graphite. In the annealing of exposed specimens, the rate of Xe^{133} escape fell off more rapidly with time than would be predicted using the diffusion model. Furthermore, the effect of temperature on D' (where $D' = D/a^2$, and a is the effective particle radius for diffusion) was much smaller for the impregnated specimens and smaller than that expected from the North American Aviation studies, suggesting again that a mechanism other than diffusion of uniformly distributed fission products from graphite is responsible for part of the gas release.

Analysis of Fuel-element-purge Technique

Although most of the effort devoted to decreasing fission-product escape is directed toward increasing retention in the fuel, it may prove to be more practical to use the alternate

approach of removing fission products from the fuel elements before they can enter the coolant system. An advanced reactor design of the United Kingdom[34] is based on this concept, as is one proposed for the High Temperature Gas Cooled Reactor (HTGCR),[35] a design by the General Atomic Division of General Dynamics Corporation in cooperation with the Bechtel Corporation, Philadelphia Electric Company, and a number of other electric power companies. One design of a fuel element which used this alternate approach consists of a graphite-uranium rod surrounded by a low-permeability sleeve. The interior of the fuel element would be held at a lower pressure than that of the coolant system, and helium coolant would be drawn inward through the sleeve. The inward flow of gas would oppose the escape of fission products outward into the coolant system.

In spite of the forced circulation of purge gas through the porous sleeve, there would be diffusion of the fission products against the flow. At low purge rates, this back-diffusion could permit the escape of significant amounts of fission gases. The influence of the flow variables and the properties of the sleeve material on the rate of fission-product escape has been examined analytically by Weissberg and Berman[36] of the Oak Ridge Gaseous Diffusion Plant. They treated the general problem of back-diffusion through a porous wall and then applied the method to a particular example of the diffusion of krypton with helium purge through a graphite sleeve.

The net rate of transport of fission gas through the wall is the algebraic sum of three component flow rates (excluding thermal diffusion), which can be expressed by the following equation:

Net flow of fission gas = ordinary diffusion
+ pressure diffusion – convective transport

"Ordinary diffusion" refers to the process of mass transfer under the driving potential of a concentration gradient. In very small flow passages, such as may exist in porous solids, collisions of gas molecules with the wall, as well as with other gas molecules, contribute to the resistance to diffusion.

"Pressure diffusion" is the term applied to the tendency of the heavier components of a multicomponent mixture to diffuse toward the higher pressure region when a total pressure gradient exists. This diffusion potential is not associated with a difference in concentration, and it exists in homogeneous as well as in nonhomogeneous mixtures. The "driving force" for pressure diffusion is a function of the concentrations and molecular weights of the components and of the total pressure gradient. The rate of mass transfer by convection is equal to the product of the concentration and the flow rate.

Weissberg and Berman solved the diffusion equations in a form which relates the escape ratio (the rate of gas escape through the sleeve divided by the rate of evolution from the fuel) to the concentration of fission gas in the entering sweep fluid, the rate of fission-gas evolution, and the sweep flow rate. Although the equations have been applied to a particular example, the results are more heuristic than valuable in themselves. As indicated by the authors, the accuracy of the numerical values is uncertain since the method of introducing the pressure-diffusion effect has not been subjected to experimental verification. In addition, the estimated escape rate depends on the effective diffusion coefficient, which could be in error by an order of magnitude. Nevertheless, the study shows the relation among the variables that influence the escape rate, and it provides a technique for estimating the escape rate which can be used if the necessary physical properties are known.

One particularly interesting result is the revelation that, for a specified void fraction and tortuosity (ratio of mean flow-path length to thickness of the sleeve), decreasing the pore size can either increase or decrease the escape ratio. This occurs because the pressure-diffusion rate increases with the higher pressure drop accompanying a reduction in pore size, whereas the ordinary diffusion rate decreases. (M. W. Rosenthal)

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Fuel-element Leak Detection

Large high-flux power reactors, especially gas-cooled reactors, are often operated with the fuel-element claddings near their critical temperatures, and, in liquid-cooled reactors, the claddings are subject to corrosive attack. Although radiation does not destroy the tensile strength of most cladding materials, fuel metals have been known to swell from the retention of fission products and to impose destructive pressures on the cladding. The detection of burst fuel elements, and their subsequent location within a large core, has therefore become an important problem of reactor monitoring systems. The design criteria for such systems are that they have reliable fast responses and that they be sensitive to small leaks so that the contamination of the coolant system by fission products will be minimized. Several methods of detecting leaks in fuel-element cladding, some of which are in the development stage, are discussed here, with reference to both liquid- and gas-cooled reactors.

A fuel element can be pressurized and a capillary tube led from each element to a pressure gauge outside the reactor shield. A drop in pressure is then a positive indication that the element is leaking. This system is employed in the Brookhaven National Laboratory reactor. Helium is used as the fill gas because of its inertness and heat-conduction properties.[1] The primary objection to this system is the difficulty in loading and unloading a large number of fuel elements fitted with capillary tubes.[2]

The coolant can be directly monitored for radioactivity, along the main coolant stream or by individual cooling channels. However, if the coolant or its contained impurities are activated within the reactor core, the energy-response band must be limited. In one heavy-water system a "double-channel monitor" utilizes pulses from a NaI scintillator placed in the coolant stream. The 5- to 6-Mev gamma spectrum for N^{16} is used as a reference for power level and background in the first channel; the second channel covers the 2- to 3-Mev band, which is composed of scattered photons and a predominant 2.9-Mev signal from I^{136}. This system has detected 80 per cent of the important failures, with some forewarning. A more sensitive system that has been tested includes a "low-energy gamma monitor," with 0.1-Mev gamma activity from Mo^{99}, Kr^{85}, Np^{239}, and Ge^{144} as the signal.[3]

Another direct-monitoring system, designed for the Submarine Advanced Reactor Prototype (S3G) reactor at the Knolls Atomic Power Laboratory, consists of a delayed-neutron counter[4] that is sensitive to leaks 1 cm square. The same sensitivity is expected from a similar system for the fast factor at Harwell.[5] The U. S. Naval Research Laboratory has tested Cerenkov detectors;[6] however, their practical performance has not yet been reported.

If the coolant delay time between core and detector is sufficient, the delayed neutrons from I^{137} and Br^{87} can be detected in a sample stream by BF_3 proportional counters or neutron-sensitive scintillation counters, as mentioned above. However, if the coolant contains oxygen, the neutron decay of N^{17} formed within the core will add to the background, as will the photoneutron decay of deuterium in light and heavy water, unless the sample delay time is sufficiently long to allow for a reduction of these events. The effect of delay time and the small yields of the delayed-neutron precursors are limitations of this system in some applications.[2]

At Chalk River, gaseous fission products are stripped from a falling film of D_2O by passing helium over it. The stripping gas is then monitored by halogen beta counters.[7] Sufficient warning of imminently serious failures is attributed to this system.

If bulk monitoring of reactor coolant streams for fission products is required, cation-exchange columns can be utilized to remove nonfission-product activities from the coolant, thus improving the signal-to-background ratio. The effluent from the cation column containing only fission-product iodine activity can then be monitored. Improved sensitivity is obtained by inte-

gration of the fission-product iodine, using anion-exchange resins. Slow response time is, however, a drawback to systems employing integration techniques. Work on systems of this type for use in Materials Testing Reactor (MTR) type reactors has been reported by Heath.[8]

A system that is becoming widely used is the "wire precipitation" method, which was developed originally at the Oak Ridge National Laboratory.[9] It is used in both liquid- and gas-cooled reactors. In both cases, sample pipes from each fuel channel are brought out of the core into matrix valves. The samples from 4 or 8 channels are combined, and the gas is filtered to remove particulate matter or the liquid is scrubbed to remove the noble fission-product gases. The effluent gases are passed through the precipitator, where the solid charged daughter products of these fission products (chiefly beta-particle emitters of iodine and rubidium from xenon and krypton decay) are collected as they are formed on a charged wire. After a sufficient precipitation time, the wire is drawn into a counter, and the activity on the wire is measured. Meanwhile, a sample from another bank of channels is being collected. If abnormal activity is observed, manual valving into a stand-by "precipitator" locates the troublesome channel. At Calder Hall, the British report[10] sensitivity to 4.1 mm^2 of exposed fuel and location within 40 min; the normal selection cycle takes 30 min. The new British power station at Berkeley will incorporate improvements in the gas-sampling system and the wire precipitator[11] and will employ complete automatic data processing for the monitoring system.[12]

This system is presently being considered for use in the Experimental Gas Cooled Reactor (EGCR) at Oak Ridge. However, the process for fission-product release is mainly that of recoil from exposed fuel; and, since UO_2 of slight enrichment contained in welded stainless-steel capsules is to be used in the EGCR instead of natural-uranium metal clad in Magnox, as in the British reactors, new problems may be introduced with respect to the required sensitivity.[13]

Reliable burst-slug detection systems are presently available for most reactor applications. However, further study is being carried on in an effort to simplify the present methods. The newer methods being considered are gas chromatography[14] and tracer-element detection.[15] (J. L. Kaufman)

Decay Heat Removal from Research Reactors

Approximately 6.9 per cent of the energy released in the fission of U^{235} originates from the radioactive decay of the resultant fission products. About 2.9 per cent of the energy appears in the form of gamma radiation, and about 4 per cent appears in the form of beta radiation. This energy, which is eventually degraded into heat, is not generated simultaneously with the fission event but appears at some later time, depending upon the decay constants of the fission products. This means that heat will be generated in the reactor for an appreciable time after shutdown.

Probably the best known attempt to estimate the magnitude of this energy release is that of Way and Wigner,[16] who found that the time dependence of fission-product energy release [in Mev/(fission)(sec)] was given, approximately, by the relation

$$E = Kt^{-1.2}$$

where t is the time since fission in seconds. For the emission of beta particles, K can be taken to be 1.44; and, for the emission of gamma photons, K is approximately 1.26. These relations hold fairly well for values of t greater than 10 sec. After long operation the heat-generation rate drops to 6.9 per cent of the operating power at shutdown, and thereafter falls off at a rate proportional to $t^{-1.2}$.

More recent estimates of the fission-product energy release have been made by Knabe and Putnam,[17] who not only have compiled data which permit an accurate estimate of the shutdown heating rate but also have compared the results obtained by several other investigators. It is important to note that none of the investigators give results which extend to the instant of shutdown. The initial times considered begin with 1 sec after shutdown.

In view of the persistence of the heat source, even though greatly reduced, it is necessary to provide an adequate heat-removal mechanism that will operate for some time after shutdown. This is true not only for the reactor but also for individual fuel assemblies that have been removed from the core. Since the heat source after shutdown is proportional to the operating power level, the size of the heat-removal system required will depend upon the previous re-

actor power. In many cases the design of the reactor system is such that the intrinsic heat removal is adequate. For example, even with the fans off and the dampers closed, there is sufficient stack-induced draft through the air-cooled ORNL Graphite Reactor to prevent a significant temperature rise in the fuel following shutdown.[18] This reactor operates at 3800 kw. The 3-Mw Low Intensity Test Reactor (LITR), which is light-water cooled, is adequately protected against damage from afterheat by natural convection of the cooling water. Natural convection of the cooling water in the Oak Ridge Research Reactor (ORR) is also sufficient to remove afterheat following short periods of operation at power levels up to 17.5 Mw. Without forced circulation of the cooling water following shutdown of the ORR after operation at power levels above 17.5 Mw, the water has been observed to boil.[19]

In the case of high-powered reactors, such as the ORR, which are cooled by forced circulation, the coastdown time of the main coolant pumps or fans is an important factor in establishing the heat-removal rate immediately following shutdown. Since the heat-generation rate decays with time, it is necessary to supply protection in the form of emergency cooling equipment only until the heat production has been reduced to the point where it can be adequately taken care of by intrinsic means. The time required will depend upon the particular system under consideration and may vary from zero to several hours. Most nuclear-power stations, for example, the Yankee Atomic Electric Company nuclear-power station,[20] employ a separate cooling system that can be operated as long as required by maintenance or refueling operations.

In sizing an emergency system, it is often possible to take advantage of the fact that the main pumping equipment will continue to operate for some time after the power has been shut off. Studies to determine the times and flow rates involved must be performed for each individual case. In a recent article, Burgreen[21] presented a method of estimating the flow velocity during a pump coastdown, and other authors[22,23] have studied the effect of various types of flow interruptions.

In a series of experiments reported by McGoff,[24] a stainless-steel subassembly for a pressurized-water reactor was inductively heated to constant power levels of 260 and 600 Btu/(hr)(cu in.). In one series of experiments the assemblies were cooled by submerging in water, and in another set of experiments they were cooled by spraying water on the surfaces. These cooling tests were conducted with the elements in inclined and vertical positions in each set of experiments. It was found that cooling was more easily achieved with the fuel assemblies in the vertical position.

Tests of an exploratory nature have been conducted by Wett[25] to determine the rate of temperature rise in irradiated ORR fuel elements held vertically in air after various decay times. The peak temperatures were found to follow the Way-Wigner relation. The results of these tests indicate that an ORR fuel element can be safely cooled by stagnant air following 20-Mw operation of the reactor after a decay period of approximately one day.

All these studies point up the conclusions that the existence of decay heat must be considered in the design of a reactor cooling system, that sufficient independent emergency cooling must be provided to prevent damage from excessive temperatures which could result from the loss of coolant flow coincident with shutdown, and that the emergency cooling may take advantage of flow coastdown following failure of pumping power and of intrinsic heat-removal mechanisms. Conservative estimates of the heat-generation rate immediately following shutdown are required. (F. T. Binford)

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Uranium-metal Fabrication Hazards

The nuclear safety problems encountered in uranium-metal fabrication are, in general, different from those involved in the production of uranium-containing solutions and slurries. The nature of the metal-fabrication process is such that the material cannot be contained within air-tight vessels during processing, and therefore criticality and health physics controls cannot be established by design specifications. Since the material is not sufficiently radioactive to require shielding for the protection of personnel, the product is generally accessible. Both health physics and criticality controls must therefore rely more completely on administrative controls than would otherwise be necessary.

Radiation Hazard

The external radiation hazard consists of the beta radiations from Th^{234} and Pa^{234}, the first and second daughters of U^{238}, which are the only decay products having sufficient energy to present a significant hazard. Persons engaged in the processing of normal or depleted uranium are usually provided with film badges, although experience has shown that it is unlikely that these individuals will receive external doses as high as the maximum permissible dose suggested by the NCRP. Since the Th^{234} and Pa^{234} daughters of U^{238} are, for all practical purposes, not present in enriched uranium, it has been found to be unnecessary to provide film monitoring in areas processing only enriched uranium.[1]

In general, the uranium radiation hazard is negligible until the material is taken into the body. However, once the material is inside the body, the hazard is considerably magnified because of the dense ionization associated with the alpha particles. Of the three modes of entry, i.e., inhalation, ingestion, and absorption, inhalation is the only one of real concern. The problems involved in controlling inhalation consist in engineering design for the control of dust-producing operations, measuring the concentration of uranium dust available for breathing, and estimating the uptake of uranium by employees in the process areas.

The magnitude of the dust-control problem is indicated by the studies of Blackwell[2] and of Hyatt[3] on the hazards incident to the rolling of uranium metal. The temperature of the metal has been found to be an important factor in defining the degree of surface and air-borne contamination. Both Blackwell and Hyatt state that, when rolling uranium at 600°C, the precautionary measures taken must be proportionally more stringent than when rolling uranium at about 300°C. Harris and Kingsley[4] have studied tests involving forging by hammer or press, extrusion, drawing, swaging, machining, and powder-pressing operations. They collected information from several uranium-fabrication plants and, in many cases, obtained comparative data which showed the air-borne contamination that existed both before and after corrective measures were instituted to reduce the hazardous health conditions.

The first step in the prevention of inhalation and ingestion of uranium is to establish control over the ventilation of process areas. The air flow is directed, in so far as possible, from the operator to the work, and natural settling of the dust is complemented rather than opposed. Conditions leading to turbulence of the ventilation air are avoided.[4]

Specific unit operations that lead to dust generation are performed in hooded areas. The hoods or dry boxes are fitted with glove ports to permit manual operations and are equipped with exhaust-air filters. Where the hoods must be quite large, as for a rolling operation, entry of personnel to the area is restricted. Employees who enter such areas are provided with protective clothing and with respirators to remove particulate matter. Grinding, pulverizing, and mixing operations are performed in closed systems whenever possible. As in any fabrication process, good housekeeping is essential; and, in

this case, it provides an additional means of exposure control in that surface contamination can be removed and settled dust can be prevented from becoming air-borne.

Monitoring Program

The effectiveness of control measures is usually assessed with several types of monitoring instruments and procedures. Air samplers are installed throughout process areas to continuously monitor for air-borne contamination. The air samplers are permanently installed 5 to 6 ft above the floor in locations which generally yield representative air samples. Samples are taken from each sampler about five times a week. In addition, portable air samplers are used for spot checks within an area and at particular machines during operations that are hazardous.

Each air sample is held for at least 16 hr after collection to permit decay of the naturally occurring short-lived alpha-emitting daughters of radium and thorium, i.e., Rn^{222} and Rn^{220}, respectively. The alpha disintegrations of the remaining long-lived uranium isotopes are then counted in an automatic scintillation counter. In most fabrication plants the established maximum permissible concentration values for occupational exposure to radioactive contaminants in air or in water are those recommended by the Subcommittee on Permissible Internal Dose of the NCRP.[5]

Since uranium contamination on surfaces in the work areas can become air-borne and thus available for inhalation, surveys are made to detect the presence of loose contamination on work surfaces. Such surveys also provide an indication of the effectiveness of housekeeping practices. Removable contamination is monitored by smear sampling. In taking smears, a piece of filter paper is rubbed over a prescribed area of about 100 cm^2 at random locations. Such locations include floors and surfaces on which dust has settled, such as pipes, window sills, and light fixtures, as well as work surfaces. All production areas are covered by such smear-sampling surveys at suitable intervals. In the enriched-uranium fabrication areas, such surveys are made weekly or monthly.

The most comprehensive program for personnel monitoring consists of periodic urinalysis. Estimates are obtained from the analyses of the amounts of activity being retained in either the lungs or the skeleton and slowly eliminated. Employees are asked to provide samples weekly, monthly, or quarterly, depending on the potential for exposure in their work area. Urine samples from persons working with normal or depleted uranium are measured by a fluorometric method; whereas those from persons working with enriched uranium or in areas where both normal and enriched material are handled are counted for alpha disintegrations following electrodeposition of the uranium onto a disk.

The possible body burden of uranium is derived from the urinalysis results by using the metabolic model of Neumann,[6] as described by Patterson,[7] and then related to standard permissible body-burden values.[5] Known cases of inhalation of significant amounts of the more soluble compounds of uranium are treated as special cases and are evaluated by considerations other than those of the routine monitoring program.

In general, the contamination problems encountered are of a technical nature. Procedures for preventing the spread of contamination and frequent monitoring of production areas are the keys to contamination control. With the exercise of such control, no hazard is encountered by personnel.[2]

Criticality Control

Criticality control is normally effected by the use of "safe-parameter" limits on the amount and configuration of the uranium metal. Such limits, as compiled from many sources, have been listed by Wachter.[8] Dry metal with less than 5 per cent enrichment in U^{235} cannot be made critical; however, criticality limits must be considered if there is a possibility that such dry metal can be flooded with water, or otherwise moderated by graphite, oils, wood, and the like. In general, limits for heterogeneous material of low enrichment must be more restrictive than those for homogeneous material because the heterogeneity results in a favorable balance of resonance escape and thermal utilization factors in latticed moderated systems. Mass limits for metal enriched with less than 5 per cent U^{235} were recently adopted by the Criticality Committee of the AEC Oak Ridge Operations Office.[9] These mass limits apply to material of an arbitrary degree of heterogeneity and moderation. At higher enrichments, criti-

cality controls for heterogeneous systems must be based on either safe geometries for metal or safe masses for solutions.[8]

With sufficiently large units of metal, having no significant reentrant sections or concavities (such as drilled holes or thick hemispherical shells), relaxations of mass limits are frequently possible, based on specific conditions of moderation, reflection, isotopic composition, and shape.[10] Such relaxations are possible, however, only when strict administrative control is established to guarantee that the necessary nuclear safeguards are in effect. A procedure used to assure mass limit controls in a fuel-element manufacturing facility has been described by Oliver.[11] The plant is divided into units, each representing one part of the fabrication process, and these units are separated physically. The different units are located in separate rooms, glove boxes, or cubicles. Each unit is then examined individually, and a quantity of material is determined which cannot possibly become critical during the given operation. This quantity is the criticality limit for the process being examined. All movements of fissile materials between different parts of the process are controlled by a single staff member, who is responsible for seeing that the critical limits are not exceeded and that the actual moving of fissile material is done safely.

As a further safeguard, double locks are installed at strategic locations throughout the plant so that any major relocation of fissile material requires the active participation of persons charged with the control of such material. All movements of fissile material are recorded, and frequent plant-wide material inventories are carried out, during which the records are checked for discrepancies and for indications that unsafe amounts of uranium may be accumulating in some part of the plant.

In determining the quantity of metal that can be handled in any unit process of the production line, geometry controls, as well as mass limits, may be invoked to ensure nuclear safety. Thus, where the reduction of UF_4 to metal results in a flat ingot of metal-safe height for the isotopic composition involved, such products can be safely stored in a coplanar fashion without mass limitations. A paper by McLendon et al.[12] traces the nuclear safety considerations involved in the handling of uranium-bearing materials from ore to enriched-metal fuel elements and includes numerous examples of the use of the safe-parameter approach in establishing and analyzing unit production operations. In general, the safe-parameter values discussed were taken from the *Nuclear Safety Guide*[10] and the shipping guide.[9] Further examples of the application of the suggested values to storage and shipping units have been given by Youngblood.[13]

The nuclear safety limits applicable to unit production operations may also be derived from the so-called "double-contingency" rule. This rule states, in essence, that a particular operation is safe if it requires the occurrence of two unrelated mechanical failures or human errors before criticality is possible. This approach differs from that of using safe parameters in that the possible accidents which might occur are estimated on the basis of the special considerations applicable to the operation under discussion rather than to accidents which are assumed to be generally applicable to every plant situation. Application of this method of analysis has been described in reports by Wachter[14] and by Mee et al.[15] The first of these reports describes the method of analysis, and the second gives the practical application of the method to the nuclear safety analysis of a high-fired UO_2 production cycle.

A further method of defining nuclear safety limits has come about through the use of special experimental tests on specific items of plant equipment. Numerous experiments relating to practical nuclear safety, which dealt primarily with regular geometric arrangements of cylinders, slabs, etc., of fissile-material-bearing solutions and metals, have been performed by Fox et al.[16] and by Paxton and Graves.[17] Such experiments have wide application in the design and evaluation of production facilities. Some new products cannot, however, be fabricated in regular geometries, and therefore experimental testing of special situations has increased. Callihan and his coworkers at the ORNL Critical Experiments Facility have described several such experiments which have led to special limits for particular unit processes. Particularly notable among these experiments are those conducted on water-flooded carbon crucibles containing large chunks of U^{235} metal and experiments regarding the interactions of metal slabs in large arrays.[18]

Handling of Scrap Material

Scrap poses particular problems in metal processing which duplicate, in many respects,

those encountered in solution processing. In both cases the nature of the material, which may be fuel turnings, combustibles, heat-treatment salts, etc., and the lack of purity make accurate mass determinations difficult. In general, scrap material must be handled by the most restrictive limits for the enrichment involved. If the shape of the uranium-bearing material is such that at least one dimension is less than about $\frac{1}{8}$ in., enriched material should be handled under the mass restrictions appropriate to solutions. Material of enrichment below 5 per cent U^{235} must be handled by the mass limits for heterogeneous systems.[9] Certain relaxations can be applied in specific cases, for example, in the salvage of heat-treatment salts, since the nuclear characteristics of the material do not favor criticality.

The problem of scrap materials was the subject of a recent conference under the auspices of the New York Operations Office of the AEC.[19] Discussions during this conference pertained to the financial considerations involved in the recovery of uranium material from various kinds of scrap; the health physics considerations, which apply both in the handling of the material and in its reduction to a form amenable to transport or processing; and nuclear safety considerations. In general, it was emphasized that scrap must be handled under rules and regulations which take account of the inability to determine the true uranium content. Such caution is particularly important when shipping solutions of low-concentration material to recovery sites. Because of its low content of uranium, a solution may be stored for a long time before being processed, and it might therefore be subjected to evaporation, freezing, precipitation, and the like, which could lead to criticality hazards not normally encountered.

In general, the transport of scrap materials poses an awkward problem: the uncertainties of uranium content require that mass or geometry criticality limits be imposed, and yet the nature of the material and the financial equity in its uranium content argue in favor of bulk shipments in inexpensive containers. The problems might be solved by concentrating scrap solutions and burning combustible scrap at the sites where generated, but the undesirable health physics problems that would be encountered might be costly. The problem of cheap containers is being worked on, and methods of modifying 55-gal drums to provide nuclearly safe containers for solutions, chips, turnings, oxides, etc., have been described. (J. W. Wachter)

Removal of Fission-product Activity from Gases

Radioactive fission products must be removed from the gas streams of reactor systems and fuel-processing plants to prevent harmful exposure of persons or costly contamination of equipment in or near the installations. The importance of this problem is increasing as the advancement of nuclear technology brings higher power reactors closer to densely populated areas.

Radioactive fission products may become dispersed in gases by design or by accident. When fission products are produced in clad fuel elements, they may be released into a containment vessel by a major accident, into the coolant by slow leakage from a few of the fuel elements, or finally from the fuel-reprocessing plant. Some reactor designs specify unclad fuel elements or circulating liquid fuels that would continuously release radioactive fission products to gas streams at relatively high concentrations.

The design of a system for removing radioactive fission products from a gas stream depends upon the physical and chemical forms and amounts of the radioactive materials and upon the degree of decontamination required. The degree of decontamination needed depends upon the fate of the gas, i.e., whether it is to be recirculated or to be discharged to the environment in an exclusion area or in a populated area.

The radioactive fission products may appear in a condensed phase in the gas stream, i.e., as an aerosol, or as radioactive gaseous impurities. The problem of removing radioactive particulates from gases has generally been approached by adapting and improving conventional aerosol removal equipment. However, the removal of the radioactive fission gases has required the development and application of special processes, the details of which have depended upon the chemical nature of the gases. The radioactive gases most frequently encountered are iodine, xenon, krypton, and argon. Under special circumstances, other radioactive gases are important, such as ruthenium tetroxide, bromine, astatine, and radon. The last two

Table IV-1 CHARACTERISTICS OF MATERIALS FOR REMOVAL OF IODINE FROM GASES

	Caustic scrubber	Heated $AgNO_3$	Silver-plated metal	Charcoal	Soda lime
Iodine-removal efficiency, %	95–99	99–99.99	90–99.9	99–99.99	99.99
Resistance to gas flow	Medium	Medium	Low	Medium	Medium
Combustion hazard	No	No	No	Yes	No
Interfering gases		Cl_2, NH_3			H_2O, CO_2
Noninterfering gases	Acids		H_2O, NH_3, NHO_3, H_2S	H_2O	

gases are not fission products, and there is virtually no information in the literature on their removal. They are, however, of interest in some phase of atomic research, and chemical considerations indicate that it may be expected that astatine can be removed in a manner similar to that for iodine and that radon can be removed in a manner similar to that for the noble gases. Bromine, a fission product, can also be removed by the iodine-removal process; and, because of its lower yield, greater chemical activity, and higher permissible concentration in air than that of iodine, it is of lesser practical concern. This issue of *Nuclear Safety* is especially concerned with the review of the literature on the removal of iodine, ruthenium, and particulate matter from gas streams.

Removal of Radioactive Iodine

Iodine is of particular importance as a radioactive contaminant in a gas stream because of its very low maximum permissible concentration in air, i.e., 3×10^{-9} μc of I^{131} per cubic centimeter of air,[5] and because of its demonstrated release from melted fuel elements.[20] The importance of iodine as a gaseous contaminant was dramatically emphasized in the Windscale accident.

Several different methods and materials have been used for removing radioiodine from gas streams: caustic scrubbers; heated silver nitrate reductors; woven silver-plated metal mesh; activated charcoal; and solid absorbers, e.g., dry soda lime. The performance characteristics of these various materials for iodine removal are summarized in Table IV-1.

Caustic Scrubbers. In this process an aqueous solution of NaOH is continuously pumped over the packing in a scrubbing tower through which the contaminated gas passes. Iodine, as well as acid contamination, is removed from the gas stream. Caustic scrubbers have been used in the Idaho Chemical Processing Plant (ICPP) for several years, with an estimated efficiency of 95 per cent for the removal of iodine.[21] The addition of 0.1*M* $Na_2S_2O_3$ to the 1.0*M* NaOH solution used in the scrubber increased the efficiency significantly in tests in which the RaLa dissolver off-gas containing I^{131} was passed at 5 scfh through 500-ml fritted-glass gas-scrubbing bottles. Other materials were tested in the same way, and they were rated in order of decreasing iodine-removal efficiency as follows: (1) heated silver nitrate, (2) scrubbers filled with activated charcoal, (3) CCl_4, (4) 0.1*M* $Na_2S_2O_3$ + 1.0*M* NaOH, (5) 0.1*M* $Na_2S_2O_3$, (6) 2 per cent NaOCl solution, (7) 1*M* NaOH, (8) dry Tygon packing, and (9) copper packing.

A caustic scrubber is provided for removing the iodine from the containment building of the ORR in the event of an accident in which fission products are released. Advance tests on a reduced scale yielded an iodine-removal efficiency of 98.4 per cent in a column 25 in. long and 4 in. in diameter packed with 1-in. Berl saddles and scrubbed with 5 per cent NaOH solution. The air-flow rate was 400 to 800 lb/(hr)(sq ft), and the inlet iodine concentration was 7 to 12 μg per cubic foot of air (0.25 to 0.4 mg/m^3). The efficiency for removal of iodine was independent of air-flow rate but increased with inlet iodine concentration.[22] The full-size scrubber for the ORR is 6 ft long, with an additional 2-ft-long demisting column 19 sq ft in cross-sectional area. It is packed with 1-in. Berl saddles and is scrubbed with a 5 per cent solution of NaOH. The full-size scrubber tower was tested at an air-flow rate of 5000 to 6000 cfm with an inlet iodine concentration of approximately 15 μg/cu ft (0.5 mg/m^3). By measuring the amount of I^{131} tracer appearing in the exit gas and in the

scrubber sump, an efficiency of 99 per cent was determined. The exit gas sample was collected in a large glass bulb, which was leached extensively to recover the sample. It was recommended that an auxiliary activated-carbon bed be used if additional iodine decontamination were required.[23] Caustic scrubbers are useful for reducing iodine concentrations in gases by one or two orders of magnitude, especially where their use is dictated by other conditions, such as the presence of acid gases; however, for higher efficiencies, other methods must be used.

Heated Silver Nitrate Beds. The performance of heated silver nitrate beds was investigated at Hanford for the removal of radioiodine from gaseous effluents from a Purex chemical-processing facility.[24] Silver nitrate supported on porous Berl-saddle column packing was selected as the best material. The beds were maintained at 375 to 400°F. Higher bed temperatures result in greater iodine-removal efficiency, but the beds must be kept at temperatures below the melting point of silver nitrate, i.e., 413°F. Development tests were conducted using radioiodine-bearing gas from the dissolver off-gas stream. Data from these tests indicated that the retention of radioiodine would be 99.9 to 99.99 per cent. The full-scale units subsequently installed operated at a retention efficiency of 99 to 99.9 per cent. The efficiency was measured by comparing samples of the inlet and outlet gas collected in caustic solution. It was found that free chlorine would release the iodine already collected and that high concentrations of NO_2 would interfere with the measurements. The recommended design conditions are superficial gas velocity, 1 ft/sec; height of packing, 8 ft; container, type 316 stainless steel.

An explosion of a silver nitrate reductor used in a Purex plant has been reported.[25] The reductor became plugged during extended operation, and it was flushed with water, with ammonia, and again with water, and then restored to service. After two weeks of operation, an explosion occurred inside the reductor, near the bottom, and destroyed it. During the normal process of dissolving aluminum-clad uranium slugs, some ammonia is produced which passes through the silver nitrate reductor. If the reductor temperature is lowered during this decladding operation, exothermic reactions occur when the reductor is restored to a normal temperature. As a result of the explosion, changes were made so that the ammonia produced during slug dissolving was either removed from the gas stream before it entered the reductor or was bypassed around the reductor. Ammonia is no longer used as a flushing agent.

Silver-plated Wire Beds. In the third iodine-removal system of current interest, the gas stream is passed over silver metal. In one arrangement, a knitted mesh of silver-plated copper ribbons of the type used in culinary scouring pads is employed. This material has been investigated rather extensively by the Harvard Air Cleaning Laboratory.[26] Efficiencies of approximately 95 per cent at concentrations of 1 to 10 mg of total iodine per cubic meter of air have been obtained in a system with a bed depth of 4.5 in. which was operating at room temperature. Typical conditions for one test are listed below:

Inlet iodine concentration	1 mg of I^{131}/m^3 of air
Activity of inlet iodine	123 $\mu c/m^3$ of air
Inlet velocity	60 ft/min
Resistance of bed to air flow	0.1 to 0.2 in. H_2O
Dimensions of silver-plated copper ribbon:	
Width	0.025 in.
Thickness	0.002 in.
Silver content of ribbon	0.5 wt.%
Packing density of bed	33 lb/cu ft

Screening tests were also run with the use of normal iodine on various metals being considered for this application, including tin, copper, and zinc, as well as on slag-wool fibers coated with KI, $AgNO_3$, Cd, Co-Sb, or Ag. Copper ribbon was effective at room temperature but showed loss of efficiency at gas temperatures of 300°C and above. Activated carbon had a very high efficiency, but it was eliminated from consideration because of the higher pressure drop.

The silver-plated copper ribbon was found to increase in efficiency at higher temperatures; but, after operation at 300°C, it was no longer effective at room temperature. The iodine-collection efficiency depends on the total iodine concentration, ranging from 90 per cent at 0.05 mg of I^{131} per cubic meter of air to 99.9 per cent

at 700 mg per cubic meter of air. The efficiency decreases to 98 per cent after 30 min of operation at the higher iodine concentrations. Decreasing the ribbon dimensions or increasing the packing density improved the collection efficiency; however, increasing the bed depth had little effect. Moisture in the air improved the efficiency, and NH_3, HNO_3, and H_2S did not interfere with iodine collection.

In similar tests at ORNL,[27] silver-plated copper mesh of the type described above was observed to have an efficiency of 98 per cent for radioiodine collection when air containing 6 mg of iodine per cubic meter was passed at 170 ft/min through an 8-in.-deep bed packed to a density of 25 lb/cu ft. It was observed that the efficiency of iodine removal increased with bed depths of up to 4 in. The efficiency for a bed depth of 4 in. was 98 per cent, and it remained relatively constant with further increases in the bed depth.

A similar material, silver-plated stainless-steel wire mesh, was tested for use in the Homogeneous Reactor Test (HRT) system.[28] Wire mesh, knitted from 0.011-in.-diameter type 304 stainless-steel wire, was electroplated with a thin layer of nickel and a 0.002-in. layer of porous silver. A bed, 6 in. in diameter and 6 in. deep, was packed with the plated mesh to a density of 29 lb/cu ft, and an iodine-removal efficiency of 99.7 per cent was obtained at 120°C. The sweep gas was dry steam flowing at 3.34 lb/min. The inlet iodine concentration was 1 ppm (1 mg/m^3). The silver-coated alundum rings that were also tested for this application were not as efficient as the silver-plated stainless-steel wire.

A pilot-plant scale test of several methods of removing radioiodine from air streams is being conducted by Arthur D. Little, Inc., Cambridge, Mass., for the Hanford Atomic Products Operation of the General Electric Company.[29] Silver-plated copper, activated charcoal, and molecular sieves are being tested under various conditions of iodine concentration, temperature, and humidity. An air-flow rate of 4000 scfm is being used.

Activated Charcoal. Activated charcoal has been used for removal of radioiodine from gas streams for several years. An iodine-removal efficiency of 99.9999 per cent was observed at Hanford in a test of a system designed for the purification of Xe^{135} for use in cross-section measurements.[30] The trap consisted of 0.54 g of 60/80-mesh charcoal in a column 4 in. high and 3 mm in diameter. The sweep gas was helium, the face velocity was 4 ft/min, the iodine concentration was 110 mg per cubic meter of air, and the charcoal temperature was 25°C. The efficiency of the system was determined by passing the effluent gas from the trap through another charcoal trap at −196°C and comparing the amounts of I^{131} collected for two runs, one with charcoal and the other without charcoal in the trap being tested. In a test with continued flow of sweep gas at a velocity of 75 ft/min at 80°C, the iodine was displaced less than 1 mm in the charcoal, showing immobility of the iodine deposited on the charcoal.

In recent tests at the ICPP,[31] inlet and outlet gas samples from the charcoal adsorption beds in the RaLa off-gas system were collected on charcoal cartridges, and gross gamma-activity measurements were made. The removal efficiencies of 90 to 98.5 per cent obtained were judged to be low because of observed losses of iodine in the sample tubing. Based on a comparison of the gross gamma activity adsorbed at the inlet and outlet gases of the main charcoal adsorber and on an observed efficiency of 99.96 per cent for the caustic scrubber and charcoal bed combined, it was concluded that the actual efficiencies were higher than those indicated.

The Harvard Air Cleaning Laboratory recently reported[26] iodine-removal efficiencies of greater than 99.9 per cent for a 1.7-in.-thick bed of Columbia 8/14-mesh activated charcoal with an inlet iodine concentration of 250 mg/m^3 and greater than 99.98 per cent for a 1-in.-thick bed of Pittsburgh Coke & Chemical Company type BPL-12/30-mesh activated charcoal with an iodine concentration of 120 mg/m^3.

Measurements of iodine removal by activated charcoal at ORNL[27] have given efficiencies of up to 99.999 per cent for Columbia 6/8-mesh SXC-type activated charcoal at room temperature in columns 8 in. long operated with an air velocity of 170 ft/min. When type CWS-6 particulate filters were used in the inlet air stream to remove radioiodine adsorbed on dust particles, the iodine concentration was found to decrease exponentially with bed depth to the limit of detection of radioiodine, i.e., less than 10^{-5} of the inlet concentration. Efficiencies were determined by evaporating elemental iodine, labeled with I^{131}, into the air stream entering the charcoal and comparing the amount of I^{131} found

downstream from the charcoal with the amount found in the charcoal. The iodine downstream was collected in series on activated-charcoal filters, carbon wool, and CWS-6 filter paper; also, in some cases, it was detected by observing a large volume of exit gas with a gamma spectrometer set on the I^{131} peak during the run. The entire system was then dismantled, and the iodine was leached and measured by a gamma-ray spectrometer. Water vapor in the air at 80 per cent relative humidity was found to have interfered only slightly with iodine removal; it changed the efficiency in a particular test from 99.9956 to 99.9936 per cent. The gas velocity had very little effect on efficiency. Coarse-grained charcoal was less efficient for the same bed depth than fine-grained charcoal, but it produced a smaller pressure drop.

A method of comparing different iodine-removal systems on the basis of efficiency and pressure drop has been developed.[27] For each material a curve of iodine-removal efficiency is plotted vs. the pressure drop per unit face velocity for various bed thicknesses. When curves of this type are compared for different materials, the best material may be selected on the basis of pressure drop for any required efficiency. Such comparisons have indicated that, for efficiencies up to 99 per cent, silver-plated copper mesh is the best material and that, for efficiencies greater than 99 per cent, activated charcoal offers the least resistance to flow. Coarse- and fine-grained charcoal filters have equal merit when evaluated in this manner.

Tests have been run on commercially available Dorex type H-42 charcoal canisters[27] (manufactured by Connor Engineering Corporation, Danbury, Conn.) with the air flowing through a $\frac{3}{4}$-in.-thick bed filled with approximately 6/14-mesh activated charcoal; the bed had a cross-sectional area of 1 sq ft. Three canisters were tested in series with a gas velocity of 25 cfm. The efficiencies of the first, second, and third canisters were 99.99, 74.6, and 14.6 per cent, respectively, and the combined efficiency of the three-stage filter was 99.998 per cent. These tests were conducted at room temperature with an iodine concentration of 0.5 mg/m^3. Iodine collectors of this type were selected for the containment-building ventilation system of the Puerto Rico Nuclear Center swimming-pool-type reactor, using single-stage canisters. The Industrial Reactor Laboratory, Inc., provides an iodine collector of a similar kind for their reactor containment building that uses three stages of canisters in the off-gas system.

Dry Soda Lime. Soda lime has been tested as an iodine collector in England.[32] Specially dried and filtered nitrogen gas was passed through a 9-cm-long 0.6-cm-diameter bed of 40/60-mesh soda lime in a copper tube coated with silicone fluid; the gas-flow rate was 1 cm^3/sec, and the face velocity was 7 ft/min. A collection efficiency of 99.99 per cent was measured by using a charcoal collector in the gas stream. The radioiodine was observed to escape after approximately 100 hr; it seems possible that a part of the collection efficiency may have been supplied by the copper container.

Efficiency Determination. The determination of efficiency of a material for removing radioiodine from gases is subject to several errors. In one commonly used method, samples of the inlet and outlet gas streams are passed through small sample collectors, such as activated charcoal or caustic scrubbers, and it is assumed that the efficiencies of the two sample collectors are the same. The collection efficiency of some materials depends upon the iodine concentration.[22,26] Reduced efficiency at concentrations corresponding to that of the exit gas sample would lead to an overestimate of the efficiency. The time variation of iodine concentration in the inlet gas stream was not reported for any of the experiments; and, if the efficiency depends upon the inlet iodine concentration, such information is needed for evaluation of the results. Another source of error arises when the radioiodine can appear as a true gas or chemisorbed on particles of dust small enough to penetrate the main collector and the sample collectors with only a small probability of stopping. To illustrate, in an extreme case where 90 per cent of the iodine is chemisorbed on such dust, 90 per cent of the iodine would pass through the collector and sampling system undetected. A measured efficiency of 100 per cent would actually correspond to an efficiency of 10 per cent. In planning experiments to determine iodine-removal efficiency, every precaution should be taken to ensure that errors such as these are minimized.

System Selection. The choice of a system for removing radioiodine from the gas stream depends upon the conditions and the requirements placed on the system. Heated reductors

would not be economical if high gas-flow rates were involved. A caustic scrubber is relatively unattractive because of inferior efficiency, complicated maintenance requirements, and problems of disposing of the contaminated caustic waste solution. Silver-plated mesh is of special interest because of very low pressure drops in applications where very high gas-flow rates are used, such as in the processing of the entire flow from a stack; however, the efficiency for iodine removal appears to be limited to 99 to 99.9 per cent. Where the highest efficiency is required, activated charcoal appears to be the best material; however, precautions must be taken to prevent or control accidental ignition of charcoal used in the presence of oxidizing gases.

Removal of Radioactive Ruthenium

Volatile fission-product ruthenium tetroxide has been observed at the ICPP in the off-gas from calcining processes at 400°C under oxidizing conditions,[33] and the conditions for stability of the volatile tetroxide have been investigated. The material either decomposes or is not formed in atmospheres that are reducing with respect to ruthenium tetroxide. At temperatures greater than 550°C, the tetroxide is decomposed.

The efficiency of removal of ruthenium tetroxide from air by a nitric acid scrubber has also been investigated at the ICPP.[34] It was concluded that a scrubber washed with nitric acid (<10*M*) at 60°C would give decontamination of greater than 99 per cent. In later work,[35] various solid adsorbents were tested to determine their effectiveness in removing ruthenium tetroxide from air containing nitric acid vapors. Three materials were tested in beds with a depth of 26 in. at superficial gas velocities of approximately 1 ft/sec. The following efficiencies were obtained for removal of ruthenium from the gas: 6/16-mesh silica gel, 99.9 per cent; kaolin packing catalyst, 98 per cent; and Corning Glass Works porous glass No. 7930, 99.6 per cent.

Ruthenium volatility appears to be of importance only in special cases where highly oxidizing materials are in contact with fission products at intermediate temperatures of approximately 400°C. It is probable that the gas-cleaning equipment required for other radioactive impurities would be effective in removing volatile ruthenium tetroxide.

Removal of Particulate Matter

The problems involved in the removal of particulate matter from gases have been extensively covered in handbooks and reviews.[36–41] Of the various air-cleaning devices available, the fiber filter is depended upon in the nuclear energy industry to remove particulate matter from gases with the utmost efficiency. The mechanisms of filter action are (1) flow-line interception, in which a particle follows the gas along a flow line and the edge of the particle is intercepted by a fiber because of its finite dimensions; (2) diffusion, in which the Brownian motion of the particle causes it to move around and increase the probability of collision with a fiber; and (3) centrifugation, in which the inertia of the particle drives it against the fiber as the flow line in the gas curves around the fiber. The probability of a particle sticking to a fiber is assumed to be constant. The important variables in the performance of a fiber filter are gas velocity, particle size, fiber diameter, and number of fibers. Diffusion is favored at low velocities since the particles spend a longer time in the vicinity of each fiber. Centrifugation is favored at high velocities. Diffusion is more effective upon small particles, and flow-line interception and centrifugation are more effective for large particles. Small fibers favor all three mechanisms, but they result in greater resistance to flow. Since the processes discussed are for individual fibers, the effectiveness of a filter increases with the number of fibers in the air stream, and thus the efficiency increases with thickness and with filter density. Considering these relations, it would be expected that there is a worst velocity for any filter and aerosol below which diffusion is more efficient and above which centrifugation is more efficient. Similarly, there is a worst particle size for filter efficiency. The locations of the minima in the plots of the filter efficiency vs. velocity of particle size depend upon the values of other variables. For example, the velocity minimum would be expected to occur at very high velocities for an aerosol containing only very fine particles since centrifugation would contribute only slightly. Similarly, the velocity minimum would occur at very low velocities for an aerosol containing large particles since diffusion contributes to only a small degree in removing particles from the gas stream.

Tests of high-efficiency high-temperature air-filtering materials of special interest for

nuclear applications have been reported.[42] Filter materials manufactured by Cambridge Filter Corporation, Flanders Filters, Inc., and Mine Safety Appliances Company were tested. Glass fibers were recommended over asbestos fibers because of easier control during manufacture. For temperatures greater than 500°F, Fiberfrax filters were recommended. It was suggested that each filter material be tested at the linear gas velocity for which it was to be used. A general set of specifications for high-efficiency filters was prepared.

Thomas and Yoder[43] have reported a filter concept which indicates that almost any desired degree of efficiency can be obtained with mechanical filters. They suggest that a critical particle size be determined for a particular type of filter and air-flow rate; larger and smaller particles will, of course, penetrate the filter more readily. The filter design, based on the critical particle size, therefore consists of sufficient filters placed in series to obtain any desired efficiency.

Tests were run at Hanford[24] with the radioactive aerosol discharged from a process ventilation system as the test medium. The efficiency of glass-fiber filters was determined as a function of filter-bed depth, packing density, and superficial gas velocity; and the resistance to gas flow was investigated for five different types of fiber. The geometric mean particle size was between 0.2 and 0.7 μ. The decontamination factor was proportional to filter-bed depth and to packing density and decreased as the gas velocity increased, being inversely proportional to the 0.4 power of the velocity through the velocity range 0 to 75 ft/min. In this range the predominant mechanisms of particle removal are flow-line interception and diffusion of the particles, as opposed to particle impingement. Higher efficiencies would be expected at higher velocities as the impingement processes became predominant. Life tests were run on these filters.

An approach to the problem of dust removal in certain British gas-cooled reactors was discussed recently.[44] Radioactive dusts must be removed from streams of CO_2 at several points in the gas-cooled reactor of the type to be installed in the Bradwell Power Station. Bypass filters will be provided for the continuous filtration of a small percentage of the main gas flow. The bypass stream entering the CO_2 dryers will be separately filtered. These filters need to have an efficiency of only 95 per cent for particles 10 μ and larger. This specification is met by cyclone tubes manufactured by Mancuna Engineering, Ltd. Relief-valve filters must be able to perform with high efficiency either at full flow or at a very low flow rate corresponding to conditions where the relief valve fails to close completely. This problem is solved by arranging for small flows of gas to pass through a ceramic or porous stainless-steel filter and for high flows to be automatically diverted through a cyclone-tube separator. Banks of cyclone tubes in parallel are encased in a pressure vessel with blowing-down connections for removal of collected dust. Eight units are used, each one being 3.5 ft in diameter and 7.5 ft high and consisting of a number of small cyclones. The operating conditions for each unit are: flow rate, 27 lb/sec; CO_2 pressure, 150 psi; temperature, 200°C; and pressure drop, 4.5 psi. Tests were conducted on a block of four cyclone tubes using a test dust, Aloxite-125, an optical grinding powder prepared by The Carborundum Company, which consisted mostly of particles less than 10 μ in diameter. The tests were conducted at an air-flow rate of 3.28 cu ft/sec with 200-g samples of dust. Efficiencies averaging 99.3 per cent were observed. Theoretical calculations have shown that the conditions of this test with air simulate the conditions expected for CO_2 in the reactor station.

The use of a venturi scrubber for removal of fission products from air has been reported.[45] Tests were run on a 20 cu ft/min gas stream yielding removal efficiencies of 90 to 95 per cent for various acid mists and for mixed fission products at concentrations of 0.1 to 0.4 mc/m^3 and for very low iodine concentrations of 2.4×10^{-4} to 7.3×10^{-3} μg/m^3, using caustic solution as the scrubbing medium. (W. E. Browning, Jr.)

Reactor Containment

The enormous hazard potential of a high-powered reactor[46] after long fuel-irradiation times requires that design concepts be oriented toward safety in every step of the route from conception to construction and operation. Most nuclear reactors are not subject to violent nuclear explosions; however, nuclear, chemical, and stored-energy releases can be sufficiently large in certain cases to cause the spread of radioactivity over large areas. The relative importance of the three types of energy release

and the radioactivity release, with respect to the hazard to the public, must be determined for each reactor. Some types of reactors are more susceptible to catastrophe in case of an accident than others. Many factors are involved in determining the hazard potential, among which are materials compatibility, stored energy in the coolant, ease of release of fission products from the point of fission, melting temperatures of the materials, power density, and mechanical design.

Each design must be subjected to consistent accident analyses in which energy release and temperature excursions are studied to determine the "driving force" tendency to disperse radioactivity in the event of the maximum credible accident. Since it is desired to protect the public and the economy, the first concern is whether the radiation doses, both during normal operation and in accident conditions, will exceed maximum permissible doses. Also of importance is the time period during which the surrounding population might have to be evacuated and agricultural land confiscated to prevent large-scale spread of radioactivity by the distribution of farm products. The maximum permissible radioactive escape from the reactor is calculated by taking into account meteorological and geological effects. Once the analytical data have been obtained, an economic balance can be established between exclusion-area costs and the cost of reducing the rate of release of radioactivity.

Barriers may be erected around the source of fission to prevent the escape of radioactivity. Such barriers should be as insensitive to accidents as possible, consistent with cost; and, if subject to fissure, several barriers should be erected in series, taking care that damage to one does not lead to damage to the others. Some specific examples of barriers in series, starting from the point of fission outward, are as follows: the crystal structure of the fuel material, if the fuel is solid; the fuel cladding; the main coolant envelope; and, if required, the separate containment building. In general, two separate barriers in series are sufficient if they fulfill the following conditions: (1) during operations, significant amounts of fission products either cannot or will not be allowed to accumulate outside either barrier; (2) the barriers are sufficiently independent that the worst credible failure of either, as well as any subsequent consequences of this failure, cannot lead to the failure of the other.[47]

It has been the practice in this country to erect a steel containment pressure vessel around each nuclear power plant built on a site that is not government-owned land. This has established a precedent which could seriously hamper the development of nuclear power as an economic source of heat and electricity. Attempts have been made to codify standards for exclusion distance and containment requirements,[48] although such attempts have been opposed by some nuclear industry representatives[49] as being premature. Since nuclear power will probably not be exploited on a large scale in the near future, it is still possible to examine each proposal individually and to determine its potential danger to the public. A guide to hazards analyses would be of immediate value, however, so that each plant could be designed to fulfill the same criteria and be subjected to consistent review. Such a guide would also orient the designer to the safety considerations early in the evolution of a design and could result in appreciable savings of effort and money in that site selection and containment requirements could be anticipated in advance.

A recent publication by Brittan[50] is an important contribution to the understanding of the general containment problem. This work is divided into four main parts: (1) the problem, (2) recommended work program, (3) work progress, and (4) U. S. experience with containment of reactor systems.

The problem is defined in two parts, i.e., within the barrier and beyond the barrier. The maximum possible energy sources within the barrier are enumerated. Mechanical loading of the barrier is classified into seven categories: (1) internal pressures arising from expansion of reactor materials from the primary envelope to the environment surrounding the primary reactor structure and bounded by the barrier shell; (2) internal pressures from chemical reactions between reactor materials and the environment; (3) internal pressures arising from the addition of fission-product heat to the gases within the shell; (4) concentrated loads from spalling of the primary reactor structure; (5) concentrated loads from missiles; (6) concentrated loads applied at contact points through items normally attached to, or in contact with, the reactor structure; and (7) other mechanical loads. Radioactive loading of the barrier and the need for accurate estimates of fission-product release within the barrier are discussed.

The discussion of the barrier and the areas beyond the barrier points out the obstacles that hamper the successful construction, testing, and maintenance of containment barriers. In Brittan's words:[50]

> The obstacles consist of inapplicability of existing codes to this type of structure, lack of designation of what constitutes maximum allowable damage to the public from leakage after an accident, accuracy limitations on initial leak rate tests, nonuniform applicability of specifications of allowable leak rates, and lack of means of proving that the leak rates do not increase in the future.

The inapplicability of the existing pressure vessel codes (see Section I of this Review) points to the futility of trying to adapt outdated laws to new and obviously different requirements. Brittan states that the development of codes tailored to cover the case of power-reactor containment shells and the writing of these codes into local government statutes would lead to more realistic design, construction, and testing. At the present time, however, the case may be that a guide would be more useful and could be more readily adapted to special requirements than a set of rigid rules written into law. A proposed code for containment shells has been drawn up by the ASA[51] (see also Section I of this Review), which could serve as the nucleus of a guide to be revised as experience is accumulated and which could eventually be adopted as law. However, the hasty creation of restrictive legal obstacles might further hamper the already slow development of nuclear power.

The report by Brittan[50] includes a well-developed recommended program and review of work in progress, as well as an excellent list of references in each of the five main areas discussed below.

> Area I: Containment Requirements. Establish containment requirements based on allowable emergency doses and deposition, on the quantity and nature of radioactive poison, and on site characteristics.

Current permissible doses for normal operation and the maximum allowable abnormal dose recommendations existing at this time are reviewed. Fission-product inventory calculations are presented, along with decay radiation power and activity as a function of time after a radiation burst or an infinite radiation time. The intensity of direct radiation from an unshielded containment shell as a function of distance and shield arrangement is given. A review of meteorology discusses Sutton's formula for atmospheric diffusion and the limitations thereof.

> Area II: Codes and Criteria. Establish or modify codes and criteria for outer barrier design construction and maintenance on the basis of unique nature of loading and utilization.

Codes and special nuclear cases of Section VIII of the Boiler Code[52] are presented.

> Area III: Proof of Containment. Determine acceptable method of proving initial and continuing containment from the standpoints of design loads and allowable leak rates.

Proof of containment is discussed, and the various leak-testing schemes currently used or proposed are critically reviewed. A rather complete discussion of the reference and absolute systems of testing is also given.

> Area IV: Loads on the Barrier. Accumulate basic information necessary to establish the mechanical and radioactive loads on the barrier under emergency conditions.

The basic energy sources studied are nuclear, chemical, and mechanical. Experiments are discussed on the chemical energy released by the reactions of aluminum and water, zirconium and water, and uranium and water at various temperatures for several particle sizes and modes of combination (sprayed, poured, explosively dispersed). Pressure loads from the expansion of steam and saturated water are discussed in some detail. Sodium-water reactions and the containment of shock waves are reviewed at length.

> Area V: Load Reduction, Design Innovations, New Applications. Devise and test proposed new methods and concepts of reducing the loading or improving the economics.

External loading and the feasibility of underground construction in Scandinavia (where it may be economical) and in the United States (where it is not) are covered.

The report[50] ends with a summary of U. S. containment experience, giving the reactor, location, power, type, shell size, plate thickness, and material. The postulated energy releases for each reactor, the shell design and test pressure, the assumed percentage of fission-product release to the container, the specific and observed leak rates, and the tests used are tabulated. The values are taken from independent hazards analyses on each reactor and show the wide variance of the assumptions, particularly

as to fission-product release and the allowable leak rate. Information sources are supplied on the following reactors: EBWR, VBWR, Dresden, Indian Point, GETR, EFFBR, EBR-II, HRT, MITR, PRTR, LPTR, NASA, Yankee, APPR-1, and PWR.

The report[50] is a comprehensive review of experience to date and is intended to serve as a nucleus for further supplements as the information becomes available. The more recent work on organic- and gas-cooled reactors has not yet been presented, and it is in these reactors that the need for containment becomes more nebulous. The pressurized-water reactors, because of their high power density and the large stored energy in the pressurized water, are rather sensitive to coolant loss or pumping failure. On the other hand, low-enrichment graphite-moderated gas-cooled reactors, because of their low power density and the nature of the coolant, can be designed so that nuclear afterheat can be removed by natural convection. Hazards do exist, of course, because of Wigner energy storage at the lower temperatures and graphite reactions at higher temperatures, but these hazards can be mitigated by proper design and operation. Gas-cooled reactors without extra containment are being built in Europe at the present time. The EGCR,[53] on the other hand, has a containment shell, primarily because of the experimental loops in which unclad fuel elements and circulating contaminated coolants can be tested. The reactor itself will not be subject to catastrophe in case of coolant loss.

Nuclear power plants, unlike coal- and gas-fired plants, need not be near their source of fuel; and, therefore, to exploit their potential to the fullest, they should be built as close as possible to the market in order to save on electrical transmission costs.[49] If this ideal is to be realized at reasonable cost, it will be necessary to develop inherently safe plants requiring a minimum exclusion area and little extra containment. (M. H. Fontana)

Plutonium Handling

Plutonium constitutes a serious hazard not only because it is a fissionable material but also because it is a pyrophoric material and the tolerances for inhalation of it are extremely low. Included in the pertinent literature on the subject are two recent publications from Argonne National Laboratory (ANL), one by Shuck[54] and the other by Steindler.[55] Shuck approaches the subject from the standpoint of the hazards associated with the handling of plutonium metal, with particular reference to the practices being employed in the new fuel-fabrication facility at ANL, whereas Steindler deals with the more general problem of laboratory-scale handling of plutonium and its compounds.

Shuck[54] discusses four different types of hazard encountered in handling plutonium, i.e., criticality, alpha emission, radiation, and fire. The criticality problem was attacked from the standpoint of "always safe mass" and "always safe geometry." The importance of effective measures to maintain these conditions in plutonium handling was emphasized by the criticality accident at the Los Alamos Scientific Laboratory (LASL).[56] Shuck quotes Henry's values[57] for critical and safe amounts of Pu^{239} as the metal and in solution, as well as the corresponding values for U^{235} and U^{233}. Shuck also mentions, in connection with the criticality problem in storage areas, a danger from the human element that could easily be overlooked, i.e., the moderating effect of the human body. The usual practice of dealing with the alpha-emission problem through the use of tight negative-pressure glove boxes is followed in the fuel-fabrication facility at ANL. Since the radiation from Pu^{239} is sufficiently soft, the facilities provided for dealing with the alpha-emission problem are usually adequate protection against the radiation hazard; but, with the advent of plutonium-fueled reactors, this situation will not always prevail, as the author[54] points out.

The fire hazard is discussed at some length because of the pyrophoric nature of metallic plutonium, and the warning system used in the fuel-fabrication facility, developed by Walter Kidde & Company, is described. It is stated that water is of no value for fighting plutonium fires and that CO_2 is of doubtful value. Dry-powder extinguishers, such as powdered silica, have been recommended, but no local experience in extinguishing plutonium fires by any means is cited. The article ends with the following statement: "Standards of safe operation must be carefully formulated and then put into practice through careful indoctrination of operating personnel. The standards must not be allowed to deteriorate through the lack of supervision or discipline."

Steindler's "Comments on the Handling of Plutonium"[55] constitutes a very worthwhile ad-

dition to the literature on this subject. Although the author[55] states that, "a discussion of this sort can neither be completely authoritative nor all inclusive," the report is the most complete and authoritative discussion of the subject that has come to the attention of this reviewer. Any discussion of the philosophy of plutonium handling will almost certainly be colored to some extent by the attitudes prevailing in the author's installation, but he seems to have succeeded very well in achieving his aim "to avoid a dogmatic approach to some of the more fundamental discussions." This report will be especially helpful to those initiating laboratory-scale research with plutonium or other high-activity alpha emitters, but it will be of interest to active workers in this field as well. Although the author has limited his discussion to plutonium handling, it should be emphasized that most of the considerations which he mentions apply almost equally well to the handling of other alpha emitters that do not require biological shielding. One must agree with the author's conclusion that objective evaluation of such factors as alternative methods of carrying out manipulations, the consequences of unforeseen incidents, and the experience and character of the personnel assigned to the work is very nearly impossible. However, although the inclusion of the urgency of the work to be performed as an important consideration may be understandable to most investigators, in the opinion of the reviewer it is questionable whether any degree of urgency would justify taking risks in handling plutonium which would not be permitted if more time were available.

A large part of Steindler's report[55] is devoted to discussions of completely closed glove boxes and techniques used in transfer operations. In this connection, it may be mentioned that LASL[58] personnel experienced in the use of the "ice-cream-carton" technique for transferring small containers of plutonium will probably not agree with Steindler's statement that this operation requires several glove changes and extensive monitoring. The 10-in. glove ports incorporated in the glove boxes of an ORNL installation[59] have been used extensively for transfers of containers sealed with the "twisted plastic" technique that is presently used at LASL. The use of glove ports for transfer operations is facilitated by the use of inexpensive gloves, fabricated locally, which consist of thin plastic sleeves connected to neoprene rubber gloves.

Open-faced enclosures and "table-top" arrangements received only cursory treatment by Steindler, apparently reflecting his belief that most operations involving plutonium are best done in completely closed glove boxes. This appears to be the feeling of most plutonium researchers, but certainly not all. The use of fluorine and vacuum systems in glove boxes is discussed, and a technique for the preparation of solids for X-ray diffraction analysis is described. Optical microscopy in glove boxes, which has been very helpful for the identification of plutonium compounds at LASL[60] and at ORNL,[61] has apparently been little used at ANL. The problems of waste disposal and storage are only briefly mentioned.

A part of Steindler's report is devoted to "Safety Control of Activity and Emergency Procedures." Methods of extinguishing and controlling fires in plutonium facilities are cited from the literature. The weak point in most glove-box systems in case of fire, i.e., the glove ports, appears to be adequately provided for at ANL by the use of metal glove-port covers attached to the inside of the plastic glove port. In any case, the best cure for plutonium fires is prevention, which the author[55] dealt with very briefly. He lists a number of desirable regulations designed to minimize hazards in glove boxes, such as keeping out flammable solvents, controlling the conditions under which perchloric acid is used, keeping sharp-pointed or sharp-edged tools and objects to a minimum, and controlling the quantity of plutonium present in glove boxes. Under the heading "Control and Detection of Contamination," the author discusses the use of protective clothing and respiratory equipment. The different types of monitoring for escape of alpha activity from glove boxes are mentioned briefly.

The severity of the plutonium containment problem is emphasized by the following tabulation of the maximum permissible body burdens and maximum permissible concentrations of plutonium isotopes in air and water for occupational exposure.[5] In all instances the critical organ (for insoluble activity) was the bone. For the most common isotope, Pu^{239}, the following values are equivalent to an air concentration, in micrograms of Pu^{239} per cubic centimeter of air, of 9.4×10^{-12} and a body burden, in micrograms, of 0.65.

	Maximum permissible body burden, μc	Maximum permissible air concentration, μc/cm³
Pu^{238}	0.04	7×10^{-13}
Pu^{239}	0.04	6×10^{-13}
Pu^{240}	0.04	6×10^{-13}
Pu^{241}	0.9	3×10^{-11}

In the papers reviewed above, very little attention was devoted to methods of providing shielding to protect operating personnel from exposure to gamma radiation, X rays, and high-energy neutrons. Several papers[62] presented at a recent symposium on plutonium fuel technology point to the fact that problems of this type must be faced in handling plutonium produced by power reactors since it will contain appreciable concentrations of Pu^{240}, Pu^{241}, and Pu^{242}, as well as Pu^{239}, and significant amounts of fission-product gamma producers. Neutrons are produced by the spontaneous fission of Pu^{240} and Pu^{242} and by the (α,n) reaction with the light elements with which the plutonium is intimately associated, as, for example, with the fluorine in PuF_3 or PuF_4. At the symposium, Bachman reported that the use of a 10-mil overlay of zinc-impregnated rubber on gloves reduced the hand exposure by 50 per cent, whereas a similar amount of lead-impregnated rubber appeared to reduce the exposure by only 60 to 65 per cent. Neutron shielding is difficult to provide since, as Bachman states, it takes about 6 in. of high-hydrogen-content material to reduce the neutron exposure by a factor of 10. Bachman also says that safety glass has been employed to remove low-energy X rays but that it will be necessary to use lead glass to minimize the exposure properly.

The dosage from high-exposure plutonium was discussed by Roesch at the symposium. His calculations showed that, with 10,000-Mwd/ton material, the hard component in the radiations, which results from the branching decay of Pu^{241} to give Am^{241} and U^{237}, increases in five days from 0.2 rad/hr to 2 rads/hr. The results of calculations of the dose rate at the surface of individual pure isotopes of plutonium when in radioactive equilibrium were also presented. From these values the surface dose rates of any mixture of plutonium isotopes can be calculated. The total gamma-ray dose of freshly purified metal rises from about 1 rad/hr for low-exposure reactor material to a little over 4 rads/hr for 10,000-Mwd/ton material. Neutrons from spontaneous fission add to the total dose rate at the surface of plutonium metal. The relative biological effectiveness (RBE) dose rate at the surface of a 1-kg sphere of metal rises from zero for low-exposure material to about 0.9 rem/hr for 10,000-Mwd/ton material, and the dose rate does not change with time after purification. The dose rate of a given distance from the surface can be calculated from the following relation:

$$\frac{\text{Dose rate at given distance}}{\text{Surface dose rate}} = \frac{\text{solid angle subtended by source}}{2\pi}$$

This equation is expected to be accurate for gamma rays and approximately correct for neutrons. It is pointed out that, as far as gamma rays are concerned, a very thin layer of plutonium will give the same exposure as an infinitely thick layer because the mean free path for gamma rays is only 0.004 in. The practical consequence of this observation is that, when plutonium is distributed as contamination on hood surfaces, the X-ray dose close to the hood is nearly equal to that on the surface of a solid piece of plutonium.

Procedures for the removal of plutonium from various surfaces were summarized in a recent report by Christensen.[63] Although the decontamination experience with plutonium is somewhat limited, Christensen reports on fission-product decontamination, which is said to be generally applicable to plutonium decontamination.

At the plutonium fuel technology symposium,[62] Wick and Thomas described the design and operation of Hanford's plutonium metallurgy facilities. Lead-impregnated neoprene gloves and lead glass are said to be adequate protection for handling materials of moderate exposure, but handling of high-exposure material will probably require remote-handling techniques or automatic fabrication processes which result in little or no personnel exposure. Design of facilities at Hanford to minimize the spread of uncontrolled contamination and to facilitate decontamination procedures includes (1) compartmentalization of the laboratory area; (2) flow of laboratory air from areas of low contamination

potential to those of high contamination potential; (3) downdraft air flow to floor exhausts below glove boxes and hood; and (4) provisions for static inert-gas or dry-air atmospheres in glove boxes, with adequate exhaust capacity to provide a minimum flow of 120 linear feet per minute through openings such as ruptured gloves and air leaks. The authors state that "thorough training of engineers and technicians in nuclear safety, coupled with operational rules, must be relied upon to maintain safe handling conditions." (C. J. Barton)

Blast Effects on Containment Shells

The unique requirements of reactor containment shells have necessitated a better understanding of the internal blast resistance of steel pressure vessels. Containment vessels designed to withstand pressures calculated by the thermodynamic equilibrium model will be overdesigned for long-time pressures, particularly if any cooling effect on the enclosed vapors is neglected, and will be underdesigned for short-term shock loading. The usual procedure of designing to the ASME code[52] gives a large safety factor that is partly intentional and partly the result of the material thickness throughout the vessel being established as that required by the stress concentrations.[64] This safety factor allows, approximately, a factor of 4 between the design loading and the yield point. Since a ductile material can be strained beyond its nominal yield point without apparent damage when loaded at a high strain rate,[65] a fairly large factor is built into containment shell design which can be considered as insurance against a possible blast. As is evident, a better understanding of the loads to be expected in cases of credible accidents and a knowledge of the behavior of pressure vessels under impact loadings would permit more economical designs of containment vessels.

Both theoretical and experimental studies of blast effects are being made. A theoretical calculation of the shock hydrodynamics of an exploding steam pressure vessel[66] was made by the Armour Research Foundation for Alco Products. In this case, a water-pressurizer vessel was assumed to detach instantly, and the mechanics of energy release and shock propagation were studied. The cylindrical pressure vessel considered in the calculations was 7 ft high and 18 in. in diameter; it weighed 4600 lb and contained 36 cu ft of saturated water at 1500 psi and 600°F. The postulated accident was an instantaneous failure of the bolting assembly by which the pressurizer was attached to its base. This failure allowed the rapid expansion of the water-steam mixture into the cylindrical vapor container, which was approximately 32 ft in diameter and 64 ft high and which contained a net air volume of approximately 34,000 cu ft at a pressure of 15 psia. It was found that the energy release from the pressure reduction, caused by the passage of a rarefaction wave through the water, created a shock problem.

The change of state upon flashing releases energy equivalent to many pounds of TNT and constitutes an explosion. The shock wave generated by the rupture is enhanced by local reflection at many surfaces and presents a danger to other components in the near vicinity of the pressurizer. The long-term static pressure was calculated to be 27.5 psia.

A characteristic of strong spherical shocks is that they propagate with a peak pressure on the shock front which is about double that of the interior pressure. Thus, in this case, a pressure of about 55 psia would exist over the entire container just before the shock wave reached the container walls. After collision of the shock wave with the vessel walls, reflected pressures would cause a pressure multiplication of the incident shock strength which would be a function of both the shock strength and the angle of incidence. For normal incidence, the incident pressure ratio and the ratio across the reflected shock (3.67 and 2.9, respectively, for this case) multiplied and yielded an overpressure of about 160 psi on the walls of the vapor container at a distance somewhat removed from the origin of the shock.

In the vicinity of the pressurizer, the reflected overpressure was estimated to be 465 psi. Local peak transient pressures of the order of 1000 psi might be reached near the pressurizer. The duration of the peak overpressure was calculated to be about 10 msec. The authors[66] categorize the seriousness of this effect by comparing the periods of the elastic modes of vibration of the shell with the positive-phase duration of 10 msec. If the period were long compared with 10 msec, the response would be less serious than the application of a 160-psi overpressure would indicate. If the period were

short compared with 10 msec, the response could be more serious than static considerations would indicate.

The conclusions of this report[66] enumerate the following points:

1. A shock problem does exist.
2. The discharge velocity of the steam from the pressurizer will be of the order of 2000 ft/sec.
3. Pressure of the order of 500 psi can be expected in the vicinity of the pressurizer when the shock wave strikes the surrounding solid walls.
4. The pressurizer will behave as a free rocket and reach a velocity of about 700 ft/sec in 0.5 sec.
5. The long-duration pressure within the vapor container will be about 27.5 psi.
6. The explosive decompression of the pressurizer will create a shock wave producing a transient reflected overpressure of approximately 160 psi on the container walls. The problem is considered sufficiently serious to warrant further examination.

Experimental work on the elastic response of containment shells to internal blast loading has been reported.[65] In the experiments, four geometrically scaled cylindrical shell models, made of steels conforming to ASTM specifications for type A-283 grade C, were subjected to internal blast loading while unsupported (i.e., suspended in air) and half-buried in the ground. The results show that geometric modeling is verified for both support conditions and that the response to dynamic loading cannot be predicted from static considerations.

It has been predicted by dimensional analysis that the scaled impulse at the peak overpressure in a blast wave is uniquely determined by the scaled distance. The scaled impulse is the impulse divided by the cube root of the explosive charge weight, and the scaled distance is defined as the distance from the explosive center divided by the cube root of the charge weight. Theoretical analyses show that a geometrically scaled model of an elastic structure subjected to an internal blast from a geometrically scaled explosive charge will produce displacement vs. time and strain vs. time histories similar to those of the prototype, with the strain amplitude unchanged but with time and displacement changed by the linear-scale factor. The experiments described support the scaling theory. The slight variations from the scaling theory may be attributed to difference in construction. It was found that the support of the lower half of a shell by the ground did not appreciably decrease the peak strains generated by the blast loading. The ground does, however, dampen oscillations of the shell, and the dampening effect increases with the size of the shell.

In many of the experiments, the maximum dynamic strains exceed the static yield strains of the shell material by factors of 1.5 to 2.0. There was no measurable permanent deformation of any of the shells. This emphasizes observations that material loaded at high strain rates will behave elastically at strain levels well beyond static yield values.

It was found that the experimentally determined circumferential stresses are approximately 18 per cent lower than the predicted values, and longitudinal stresses are approximately 12 per cent lower than predicted. Comparison of dynamic loading effects with static pressure effects emphasizes the differences between the static and dynamic responses of the structures. Dynamically loaded shells exhibit strains oscillating about the zero strain level. Also, the relative values of the strain components at a given location differ from those for a statically loaded shell.

The results presented above apply to vessels loaded within the elastic range. Experiments on strain-energy absorption of materials loaded in the plastic range were reported previously in *Nuclear Safety*.[67] The work cited,[68] in which dynamic dilation was effected by solid propellants, showed that the strain energy absorbed in dynamic dilation of type 304 stainless-steel test specimens was somewhat larger (about 20 per cent) than the energy absorbed in static dilation. The strain-energy absorption characteristics are a function of both the constituents of the material and the prior history of the specimens. Such factors as cold work and irradiation damage could have considerable effect on energy absorption.

These experiments, and further work on both the mechanics of the internal loading and response of blast-loaded structures, should lead to more confident analysis and design of containment shells for use in situations where shock loading is possible. (M. H. Fontana)

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Atmospheric Dispersion

By FRANKLIN A. GIFFORD, Jr.*

One of the chief sources of uncertainty in estimating the hazard associated with accidental or planned release to the atmosphere of fission-product activity has been the lack of reliable measured values of atmospheric dispersion coefficients. In the absence of any obvious alternative, Sutton's well-known mathematical dispersion model[1] has been used in many reactor hazards analyses for evaluating effects far beyond the limits for which the model can confidently be expected to be reliable, e.g., distances of the order of 1 km and near adiabatic (neutral) conditions of atmospheric stability. Consequently, the appearance, in several recent papers, of a sizable quantity of new atmospheric dispersion observations is of considerable interest in connection with the meteorology of nuclear safety problems. Furthermore, the calculation of atmospheric dispersion by the method of moving averages, as has been proposed recently, seems to provide an improved means of calculating dispersion, not only because the technique has less restrictive boundary conditions but also because it is well adapted to the interpretation of continuously monitored atmospheric data.

Recent Dispersion Observations

The recent atmospheric dispersion observations differ to some extent as to the methodology, the ranges of significant variables covered, conditions of the observations, and other factors. The conditions of some recent experiments[2–7] are described in Table V-1, and some of the significant differences may be noted.

The dispersion data from these experiments have been presented in either of two related forms, depending on the author's purpose and preferences. Thus Stewart et al.[2] and Bowne[7] relate their results to Sutton's dispersion formula:

$$\chi = \frac{Q}{\pi C_y C_z u x^{2-n}} \exp\left[-x^{n-2}\left(\frac{y^2}{C_y^2}+\frac{z^2}{C_z^2}\right)\right] \quad (1)$$

where χ is concentration in grams or curies per cubic meter; Q is source strength in grams or curies per second; u is mean wind speed in meters per second; and x, y, and z are coordinate distances from the source in meters, with the x direction coinciding with that of u. The remaining terms, i.e., n, C_y, and C_z, are the parameters to be determined from the experimental dispersion data. The units of the diffusion coefficients, C_y and C_z, are meters to the $n/2$ power, i.e., $m^{n/2}$. Presentation of observations in this frame of reference is quite useful, particularly in support of reactor hazards studies, since Eq. 1 and related dispersion formulas have been a mainstay of hazards studies for a number of years and have gained wide general, if uncritical, acceptance, especially among persons other than meteorologists whose work has brought them into contact with reactor hazards problems.

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Table V-1 BASIC INFORMATION ON RECENT ATMOSPHERIC DISPERSION EXPERIMENTS

Investigator	Stewart et al.[2]	Hay and Pasquill[3]	Cramer et al.[4,5]	Hilst and Simpson[6]	Bowne[7]
Distance range of sampling, meters	150–10,000	100	50–800	150–1500	150–3400
Terrain	Built up to 6000 meters, becoming open country	Gently rolling downland	Flat prairie	Flat desert	Flat desert
Source height, meters	61	2	0.46	65	46
Tracer material	Ar^{41}	Lycopodium spores	SO_2	Fluorescent pigment	Uranine dye
Sampling interval, min	40 (15 to 60)	3	10	60	30
Sampling height, meters	Surface to 1000	0.60	1.5	65	1
Number of experiments	88	8	70	5	16

Table V-2 EXPERIMENTAL DIFFUSION COEFFICIENTS DERIVED BY STEWART ET AL.[2] FROM BEPO MEASUREMENTS

Type of survey	Meteorological conditions	Sampling range, meters	Actual sampling time, min	Diffusion coefficients,* $m^{n/2}$	
				C_y	C_z
Vertical	Unstable	150–1000	15–20	0.16	0.27
	Adiabatic			0.10	0.22
	Stable			0.10	0.12
Ground	Adiabatic and unstable	590–620	40	0.46	
		880–1050	40	0.28	0.32
		1200	40	0.33	
Ground	Adiabatic and unstable	2400–2800	15	0.26	
		6000–9700	15	0.18	0.19
Ground	Stable	1000–10,000	60		<0.04

*All values are relative to a standard sampling time of 40 min.

On the other hand, Cramer[8,9] relates the important Project Prairie Grass dispersion data to the following interpolation formula:

$$\chi = \frac{Q}{2\pi\sigma_y\sigma_z u} \exp\left[-\frac{1}{2}\left(\frac{y^2}{\sigma_y^2}+\frac{z^2}{\sigma_z^2}\right)\right] \qquad (2)$$

which is a direct consequence of the requirement of mass continuity and the assumption of Gaussian concentration distributions. With this formula the data reflect directly the spatial variability of the dispersion coefficients σ_y and σ_z. Data of Hay and Pasquill,[10] as well as discussions in references 5 and 8, indicate that, as a practical matter, the Gaussian assumption is acceptable; but a careful analysis of the Project Prairie Grass data by Barad and Haugen[11] has indicated that significant departures from Gaussian concentration distributions do occur. Clearly, dispersion data based on Eq. 1 may be related to data based on Eq. 2 through the following identities:

$$\sigma_y^2 = \frac{1}{2}C_y^2 x^{2-n} \qquad (3a)$$

and

$$\sigma_z^2 = \frac{1}{2}C_z^2 x^{2-n} \qquad (3b)$$

(A factor of 2 is absent from the denominator on the right-hand side of Eq. 1 because this equation is conventionally multiplied by 2 to account for "reflection" of the dispersing particles by the ground plane.)

The dispersion coefficients C_y and C_z, derived by Stewart et al.[2] from the BEPO measurements, are summarized in Table V-2. The C_z values were obtained from the measured

standard deviation of the concentration distribution at the vertical plume centerline, which was calculated by using Eq. 3a; whereas the C_y values were obtained from the maximum value of the concentration distribution, χ_p, through the relation

$$C_y = \frac{Q}{\pi \chi_p C_z u x^{2-n}} \qquad (4)$$

In obtaining these values it was assumed that $n = 0.25$ for all stability conditions, an assumption that has a number of virtues, not the least of which is simplicity. The work of Barad and Haugen[11] indicates that power-law formulas, such as Eqs. 3a and 3b, for the variation of dispersion with downwind distance, will have to take into account different n values for the y and z directions, i.e., for lateral and vertical diffusion. This analysis also shows that neither the n value for the y direction, n_y, nor the n value for the z direction, n_z, is related to the power-law representation of the low-level wind profile. In these circumstances it seems quite appropriate to evaluate C_y and C_z over various distance ranges, holding n fixed.

The BEPO data have been corrected for the finite size of the stack by an upwind zero-point displacement of 56 meters. The authors[2] comment on the high values found for C_z (relative to those for C_y), even under stable conditions, and attribute them to the mode of discharge from the stack. The initial rise ranges from about 80 to 135 meters above the stack top under these particular discharge conditions and is inversely related to wind speed. It is speculated that a decrease in the value of both C_z and C_y beyond distances from the source of 6000 meters reflects adjustment of the rate of diffusion to conditions appropriate to the open country beyond the built-up site area.

A summary of the BEPO results is presented in Fig. 1. The peak (centerline, ground-level) concentration values and the lateral dispersion values apply to the conditions $u = 5$ meters/sec, as measured at a height of 1 meter and a sampling time of 40 min. The authors[2] state that "the (ground-level) concentration at any distance may be greater than the plotted figure by a factor of 1.5 under daytime conditions, while the lower limit (in stable conditions) is zero. The value of σ_y is subject to a variation of ±25 per cent about the mean curve." This estimate that the peak-to-average concentration

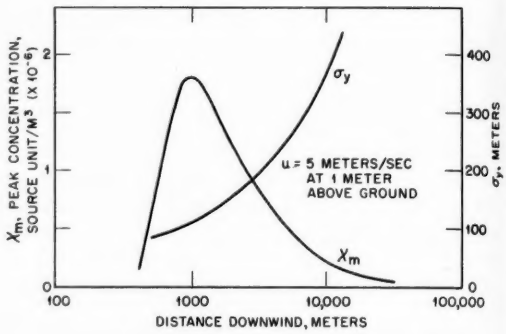

Figure 1—Average ground-level values of axial concentration, χ_m, and lateral dispersion, σ_y, for the BEPO data.[2] (Reprinted by permission from the *International Journal of Air Pollution.*)

ratio is 1.5 at all distances can be questioned. For locations near a stack base, ground-level concentration peaks of over 20 times the mean surface concentration level have been reported, e.g., by Fuquay.[12] The value 1.5 is probably valid anywhere beyond distances from the source corresponding to the maximum point of the concentration curve of Fig. 1.

Bowne's dispersion measurements,[7] which were made with the use of the recently developed uranine-dye technique of Robinson et al.,[13] have been summarized in terms of Sutton's coefficients; the pertinent C_y and C_z values are shown in Table V-3. In this study the n values

Table V-3 EXPERIMENTAL DIFFUSION COEFFICIENTS OBTAINED BY AVERAGING DATA GIVEN BY BOWNE[7]

Assumed value of n	Value of u at 140 ft, meters/sec	Average diffusion coefficients, m $^{n/2}$		No. of runs
		C_y	C_z	
0.20	8.8	0.44	0.25	10
0.25	8.3	0.40	0.15	6

were determined from low-level wind profiles and then rounded off to the nearer of the two values, 0.20 or 0.25. These experiments were performed at the National Reactor Testing Station (NRTS) in Idaho, where the locale is flat desert, covered with sagebrush, and is relatively free from large built-up areas. The conditions of release approximated a point source fairly closely, and no correction for an initial rise above the release point was mentioned. In con-

trast with the BEPO experiments, C_y was determined from Eq. 3a after the horizontal plume concentration distributions had been measured. The value of C_z was determined from the distance downwind, d_{max}, of the maximum concentration value, χ_{max}, from the formula

$$d_{max} = \left(\frac{h^2}{C_z^2}\right)^{n-2} \quad (5)$$

where h is the height of the source above ground level.

It is widely held that Sutton's dispersion model does not apply to very stable conditions. Since these are just the conditions of least atmospheric dispersion, measurements of σ_z, such as those by Hilst and Simpson,[6] are of particular interest. Their data are summarized in Table V-4. The observations were made at Hanford, where the terrain is quite similar to that at NRTS. Comparison, by means of Eq. 3b, of these values with the C_z values obtained from the BEPO measurements (Table V-2) for open country (<0.04) confirms the very slow rate of vertical dispersion applicable to stable conditions.

Table V-4 EXPERIMENTAL VALUES OBTAINED* BY HILST AND SIMPSON[6] FOR THE VERTICAL DIFFUSION COEFFICIENT σ_z^2

Test	σ_z^2, m²			
	At 500 ft	At 1000 ft	At 2500 ft	At 5000 ft
A	25	60	112	
B	24	55	93	125
C	30	79	186	245
D	30	66	152	220
E	17	36	43	42

* At various distances from a source 185 ft above ground level under stable conditions.

The Project Prairie Grass dispersion data have been summarized from the standpoint of both Eq. 1 (reference 11) and Eq. 2 (references 8 and 9). A comparison of the results with Eq. 1 emphasized that a modification was required to provide a successful fit to the data, namely, the introduction of n_y and n_z. For this reason, the Project Prairie Grass results, as summarized with respect to Eq. 2 by Cramer,[9] are shown in Figs. 2 to 4, which give a concise summary of this important and extensive series of dispersion experiments. It should be pointed out that extrapolation of the curves beyond dispersion distances of 800 meters, the limit of the Project Prairie Grass data, is to be regarded as a suggestion only.[5]

Calculation of Atmospheric Dispersion by the Method of Moving Averages

A particle of smoke or other effluent material, or a small "piece" of air, somehow marked or identified, will be displaced a lateral

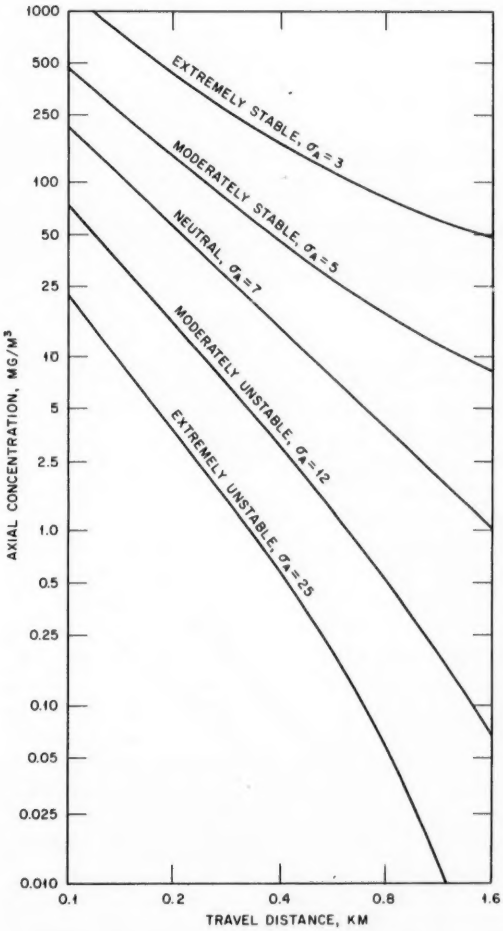

Figure 2—Estimates[9] of ground-level axial concentration values as a function of atmospheric stability based on the Project Prairie Grass data. Concentrations are adjusted to a source strength of 100 g/sec and a wind speed of 5 meters/sec; σ_A is the approximate standard deviation of the horizontal wind direction, in degrees, for each stability condition. (Reprinted by permission from the *Bulletin of the American Meteorological Society*.)

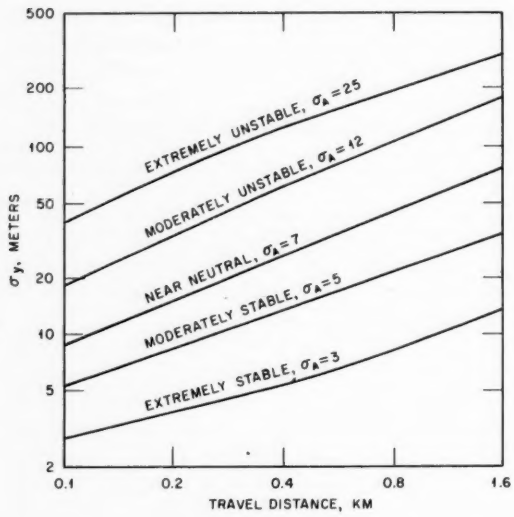
EXTREMELY UNSTABLE, $\sigma_A = 25$
MODERATELY UNSTABLE, $\sigma_A = 12$
NEAR NEUTRAL, $\sigma_A = 7$
MODERATELY STABLE, $\sigma_A = 5$
EXTREMELY STABLE, $\sigma_A = 3$
σ_y, METERS
TRAVEL DISTANCE, KM

Figure 3—Estimates[9] of σ_y for data of Fig. 2. (Reprinted by permission from the *Bulletin of the American Meteorological Society.*)

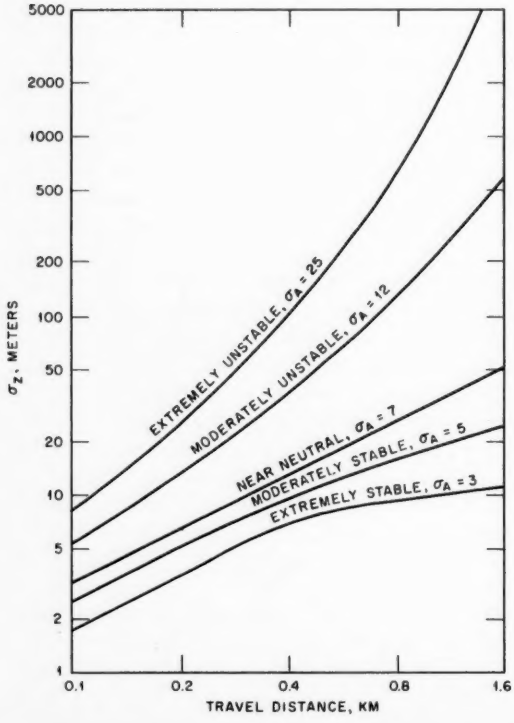
EXTREMELY UNSTABLE, $\sigma_A = 25$
MODERATELY UNSTABLE, $\sigma_A = 12$
NEAR NEUTRAL, $\sigma_A = 7$
MODERATELY STABLE, $\sigma_A = 5$
EXTREMELY STABLE, $\sigma_A = 3$
σ_z, METERS
TRAVEL DISTANCE, KM

Figure 4—Estimates[9] of σ_z for data of Fig. 2. (Reprinted by permission from the *Bulletin of the American Meteorological Society.*)

distance y, after a travel time T, under the influence of air turbulence fluctuations $v(t)$ in the y direction according to the relation

$$y(T) = \int_0^T v(t_1)\, dt_1 \tag{6}$$

The air is regarded as a fluid continuum having homogeneous turbulence properties. Squaring both sides, taking the average over many repetitions of the experiment, and rearranging give the familiar one-dimensional mean-square particle displacement

$$\overline{y^2}(T) = \overline{2v^2} \int_0^T \int_0^{\tau} R(\xi)\, d\xi\, d\tau \tag{7}$$

of Taylor's theory of diffusion by continuous movements. Sutton's dispersion model follows from the introduction of a particular form for the Lagrangian single-point autocorrelation function, R. The form of R assumed by Sutton then reduces the problem of dispersion to the determination of diffusion parameters which depend on the turbulence structure. In practice, definite scale limitations must be observed, and there are certain other theoretical and practical shortcomings that are fairly well recognized.

It was first observed by Kampé de Fériet[14] that Eq. 7 can be written, in terms of the Lagrangian frequency-power spectrum of the eddy energy, $F(n)$, as follows:

$$\overline{y^2}(T) = \overline{v^2} T^2 \int_0^{\infty} F(n) \left(\frac{\sin \pi n T}{\pi n T}\right)^2 dn \tag{8}$$

Hay and Pasquill[3] have recently pointed out the similarity of the integrand in Eq. 8 to the expression by means of which a computed turbulence-power spectrum is corrected for the effect of averaging the raw data over a time τ, as discussed in a number of references (e.g., reference 15); namely,

$$F(n) \left(\frac{\sin \pi n \tau}{\pi n \tau}\right)^2 = \overline{F}(n) \tag{9}$$

where $\overline{F}(n)$ is the "observed" spectrum. Because F is normalized, it follows that it is possible to write Taylor's dispersion equation in the following extraordinarily simple form by choosing the averaging time in Eq. 9 equal to the dispersion time (time of travel):

$$\overline{y^2}(T) = \langle \overline{v^2(t)} \rangle_{\text{av}}\, T^2 \tag{10}$$

where the symbol $< >_{av}$ indicates that the original velocity signal, $v(t)$, is first subjected to a running mean (moving average) over an interval equal to the travel time, T. In fact, this expression had been derived independently by Ogura,[16,17] directly from the definition of a moving average, in connection with his earlier studies of dispersion and the relation of space and time averaging; but recognition of its great potential utility in practical atmospheric dispersion work must be credited to Hay and Pasquill.[10]

Equation 10 says that, to get a measure of the dispersion over a time T, one simply computes the mean-square deviation of the velocity signal, $v(t)$ (which, however, has first been subjected to a moving averaging over the dispersion time, T) and then multiplies the result by T^2. Needless to say, it is necessary to have measurements of $v(t)$ over a period considerably longer than the travel time, T. Equations 7, 8, and 10 are completely equivalent. They relate the dispersion alternatively to three statistics: the autocorrelation, the power spectrum, and the variance of a running mean. The statistical reasons why computation of the power-spectrum function is generally to be preferred to that of the autocorrelation function have been emphasized in a recent monograph by Blackman and Tukey.[18] The conclusion is that the power spectrum of a sample of a stationary random function is the better behaved of the two statistics. Since either function enters, by Eqs. 7 and 8, into the expression for the dispersion in a highly smoothed form, details of their shapes are not likely to be important from the standpoint of dispersion. Consequently the readily computed running-mean statistic seems to be a promising alternative to either the power spectrum or the autocorrelation as a measure of dispersion.

It would be very useful, in practice, if the Lagrangian wind-fluctuation statistic in Eq. 10 (i.e., the variance of the moving averaged velocity of a single fluid particle) could be related to the Eulerian kind of wind-fluctuation measurements made at a single point (as by an anemometer or a wind vane). Then, with an ordinary anemometer, comparatively simple wind-fluctuation measurements could be interpreted in terms of dispersion. The problem of relating Lagrangian (fluid attached) to Eulerian turbulence statistics is extremely difficult. A recent proof by Lumley[19] that single-point Lagrangian and Eulerian statistics (such as means and variances) are, in fact, equivalent for the case of homogeneous incompressible-fluid turbulence is an encouraging theoretical development.

As a practical matter, Hay and Pasquill[10] suggest that the fluid-attached Lagrangian time scale, ξ, be identified with the fixed-point Eulerian time scale, t, by a simple proportionality, $\xi = \beta t$. Then

$$R_L(\xi) = R_E(t) \tag{11a}$$

and

$$F_L(n) = \beta F_E(\beta n) \tag{11b}$$

(The subscripts denote the Lagrangian and Eulerian time systems.) This permits writing the dispersion equation in the following form, which is fully equivalent to Eqs. 7, 8, and 10 and is certainly the ultimate in simplicity:

$$\sigma_\rho^2(T) = \overline{\theta'^2_s} \tag{12}$$

Here $\sigma_\rho^2(T)$ is the variance (after a time of travel, T) of the particle spread (in angular measure) from a continuous point source and $\overline{\theta'^2_s}$ is the variance of wind-direction fluctuations as measured at a fixed point and subjected to a moving average over time intervals $s = T/\beta$. A value $\beta = 4$ is suggested, based on their experimental data,[10] which further agrees, for example, with the result of another investigation.[15] The computed dispersion value is quite insensitive to fairly large assumed variations in β, i.e., between 1 and 10. The full utility of the method of moving averages is realized when the wind-fluctuation data are processed by electronic means. Jones and Pasquill[20] describe a circuit, employing conventional electronic techniques, by means of which the wind-direction signal from a sensitive vane is continuously filtered in a form closely approximating the right-hand side of Eq. 12.

This method of moving averages has two striking advantages. First, it relates the actual turbulent wind fluctuations directly to the dispersion, without introducing an intermediate hypothesis. As a result, there does not seem to be any a priori restriction to particular scales of dispersion time or distance or to particular meteorological conditions. This point will, of course, have to be verified by suitable

studies. Second, the method is well adapted to the continuous monitoring of atmospheric dispersive capacity at a reactor site and would appear, at least from the reactor hazards standpoint, to solve the problem of processing large amounts of wind-fluctuation data in connection with reactor-site meteorological programs, such as those described in references 21 and 22, to cite but two recent examples.

(Franklin A. Gifford, Jr.)

Biological Contamination and Dispersal of Radioactive Wastes

Hazards resulting from contamination of crops and other constituents of the biological environment may limit the scale of advances in nuclear technology in populated regions or may require safeguards which would increase costs. Evaluation of these hazards will require the formulation of a coherent theory of movement of radionuclides and stable chemical elements in complex ecological systems (ecosystems) and the application of this theory in an industrial context.

Considerable effort has already been expended in the exploration and evaluation of the amounts and rates of escape of wastes under a variety of soil, hydrologic, climatic, and other conditions; and this research has involved a wide diversity of talents covering many scientific disciplines. Attention is directed to the recent four-volume publication[23] of the hearings before the Joint Committee on Atomic Energy on industrial radioactive waste disposal. These volumes present a review of present and proposed waste-disposal techniques in the United States. One of the conclusions reached is that it will always be necessary to use the diluting power of the environment to some extent in handling huge volumes of low-level waste. A fundamental understanding of ecosystems is needed to assess the consequences of such a practice and to provide a basis for establishing long-range precautions to assure protection of man and his environment. Although ecological studies are being carried on at several institutions, the programs are in the preliminary stages of development. The reports discussed here are concerned with individual links in the chain of movements of radionuclides. Present knowledge does not yet provide a comprehensive picture of their possible role and implications in whole ecosystems.

Contamination of Vegetation

Some information on individual stages of movement of radionuclides can be gleaned from studies of fallout.[24] The comprehensive summary by Nishita and Larson[25] of both the Nevada fallout surveys and related Operation Greenhouse experiments provides a convenient point of departure for discussions of several recently published and unpublished results.

Although bomb fallout travels higher and farther than stack fallout, the results were similar in showing the sedimentation of coarse particles near the source and finer ones farther away. Vegetation served as a selective trap for fine particles (especially of silt size, less than 50 or 60 μ, which settle at velocities that are negligible in comparison with atmospheric turbulence). Near the source the total surface radioactivity was high, and the particles were large; 7 miles from Ground Zero (GZ) the particles were of medium sizes between 100 and 1000 μ, i.e., fine to coarse sand. The percentage of activity intercepted by clover and wheat near the source was only 0.1 to 0.2 per cent. At 106 miles from GZ, where most of the fallout was in the silt range, clover and wheat intercepted as much as 25 per cent of the fallout.

Direct adsorption on foliage is especially important for the short-lived I^{131}. Iodine was rapidly incorporated inside the plants, where the source was experimentally controlled to assure that it was gaseous; but over 90 per cent remained on the outside of the leaves in the case of particulate iodine contamination from an experimental fuel-element burn-up at Dugway Proving Grounds.[26] Stack contamination, which is initially gaseous, presumably becomes adsorbed rapidly onto particles. Even where internal contamination of the vegetation may be as low as 5 per cent, the external particulate contamination is very difficult to wash from surfaces of vegetation and from food products, except those with smooth waxy surfaces.

The reviewer's unpublished gamma spectra of vegetation at Oak Ridge indicate measurable dispersal of I^{131} (5 $\mu\mu$c per gram dry weight) 8 miles in the downwind direction, although the highest activity (583 $\mu\mu$c per gram) was localized within 0.1 mile of the stacks. By contrast, Ce^{144}, Co^{60}, Cs^{137}, Zr-Nb^{95}, and Ru^{106} exceeded the levels of regional fallout mainly within a

small area 0.1 to 1.0 mile around the stacks. A subsequent incident at ORNL in November 1959 (to be discussed in the next issue of this Review) involved the dispersal of Ru^{106} on coarse rust particles which also settled near the stacks. It is thus evident that normal and abnormal contamination must be taken into consideration in the planning of new installations. Specific identification of the radionuclides is essential because they differ so much from one another in their behavior in the atmosphere, biosphere, and soil.

Oak Ridge observations[27] also confirm another finding of research in the desert,[25] i.e., that there are great differences among species in their retention of fallout, depending on anatomical characteristics, such as leaf hairs and glands. Only after standardizing on certain species and collecting leaves well exposed on many parts of the crown was it possible to take full advantage of vegetation as a natural collector for monitoring contamination.

The Nevada tests[25] indicated that 75 per cent or more of the aerial contamination reached the soil directly. The fact that long-lived fission products adsorbed on the leaves will also move onto the soil from dead litter explains the emphasis placed on the uptake of radionuclides from soil.

The direct uptake of radionuclides from soil by vegetation offers no simple means of decontaminating an area.[25,28] The first and nine successive croppings of ladino clover removed only 4 and 23 per cent of the Sr^{90} and 0.13 and 0.72 per cent of the Cs^{137}, respectively. Several other crops grown to maturity removed only 9 per cent Sr^{90}, 0.5 per cent Cs^{137}, 0.13 per cent Ru^{106}, 0.03 per cent Y^{91}, and 0.02 per cent Ce^{144}. The low percentage of uptake might seem to minimize the hazard, especially for the dilute contamination by fallout; and yet these percentages, or even lower ones, are sufficient to provide strongly contaminated vegetation in areas which have been affected by low-level radioactive wastes and which have levels of contamination thousands of times as high as those which are widespread from fallout.[27]

The low uptake of Cs^{137}, reported in these[25,28] and other studies, presumably indicates fixation of a large percentage of the cesium ions in clay lattices of the soil. Instead of being depleted, along with potassium, as the vegetation was cropped several times, Cs^{137} in the plants remained at the same level for a number of croppings and then increased.

A simple interpretation might consider the increased uptake of Cs^{137} as a result of decreasing competition by potassium for available ion-exchange sites on the roots. Another complication of cesium-potassium relations in certain cases might be the influence of potassium directly on the clay structure which was fixing cesium. At Oak Ridge,[29] experiments with the extraction of cesium in solution have suggested that a certain level of potassium concentration tended to collapse the lattice of vermiculite minerals and fix more cesium between the plates than did potassium concentrations either higher or lower. Such fixation might also make cesium less available to plants and thereby provide a mechanism other than ionic competition by which potassium could modify the uptake of cesium.

In a recent set of Japanese chemical analyses,[30] the failure of the natural cesium content of plants to correlate as closely with potassium as does rubidium (coefficient of correlation r = 0.18, as compared with 0.72) provides a different indication that cesium is not following potassium as a simple "impurity." Part of the individuality of different ions is controlled by plants; cucumber, rice, and soybean have different distribution factors for the plant ratio of cesium to potassium as compared with the ratio in the nutrient solutions. However, some of the scatter in the cesium-potassium correlation might conceivably be explained in terms of differences in soil; that is, illitic minerals having extremely high ability to fix cesium would lead to a very low plant content, even though potassium might be relatively abundant, whereas certain other soils might show far less restraint of this kind. Such speculations suggest the need for analyses of closely related samples of soil and vegetation in ecosystems which are being studied intensively.

Analyses of contents of natural cesium in vegetation are still so scarce that the analytical feat of accomplishing these with a cathode-layer arc spectrograph in plant samples is noteworthy.[30] One of the authors has more recently been using activation analyses and radiochemical separations to get even better results.[31] A recently prepared review of activation analyses and conventional analyses for the determination of trace elements in marine

organisms includes a summary of many data and a bibliography.[32] It is hoped that the new analytical techniques, combined with improved concentration by ion exchange, will provide much new information on the natural amounts and movements of cesium to complement the standard data on radiocesium and total potassium.

In Sweden the addition of cesium carrier to soils was very effective in releasing fixed Cs^{137} and in increasing its uptake by plants.[33] This seems to be consistent with the idea that there are special sites in the mineral lattice where cesium is normally fixed and that there are only limited numbers of these sites which can readily be saturated by additions of carrier. In contrast, additions of potassium suppressed cesium uptake in Swedish soils, as in California soils. It is not clear whether this suppression involved simple suppression or a more specific mineralogical influence similar to that mentioned above.[29]

Returning to the strontium problem, the Swedish study confirmed the effectiveness of decreasing strontium uptake by the addition of calcium for acidic soils whose exchange complex is not already saturated with calcium. Such marked suppression was not noted for calcium-saturated soils.

Extensions of earlier work at Hanford also confirm that acidic soils show strongest suppression of Sr^{90} uptake by calcium and that this effect is not due merely to sulfate or nitrate anions in calcium fertilizer nor merely to dilution of Sr^{90} by increased yield.[34] Strontium-90 as phosphate, carbonate, hydroxide, fluoride, oxalate, or sulfate was less available to plants than as chloride or nitrate, but differences in uptake were far less in acidic Cinebar silt loam than in calcareous Ephetra sandy loam or Milville loam.

Significant double-tag experiments showed that uptake of both Ca^{45} and Sr^{85} was suppressed similarly by increasing stable calcium. However, if the total stable calcium and strontium were maintained at 2 millimoles/liter, very different ratios of these stable elements had little influence on the uptake of either tracer. This casts doubt on the contention that impurities of strontium in calcium fertilizers are chiefly responsible for the suppressed uptake of radiostrontium by plants, i.e., by simple isotopic dilution by strontium. Yet certain differences in movement of the two tracers indicate that they are not equivalent as tags for total calcium or for calcium plus strontium; each element has an individuality of its own.

Further physiological experiments, in addition to chemical and meteorological studies, will provide an approach to an understanding of the mechanisms of movement of radionuclides. A complementary approach must be field research on the diverse combinations of natural biological and environmental parameters which together constitute the boundary conditions limiting these mechanisms. Combining these approaches and the separate links in the chain of movements and effects should lead toward the fuller picture of environmental contamination which is needed as one guide for managing nuclear industries. (J. S. Olson)

Dispersal by Insects

From the beginning of the atomic industry, the use of basins or other restricted aquatic environments for the disposal of radioactive wastes has prompted concern with regard to the possible avenues of escape of these materials. A number of programs are under way which are concerned primarily with the accumulation of radionuclides by the components of the biota.[27,35,36] Singularly lacking in most of this effort is research on the role of biological agents in the dispersal of radioactive contamination.

Various workers have speculated on the possible role of algae in the decontamination of wastes in basins. The basis for these conjectures is the well-known ability of freshwater algae to accumulate radionuclides in concentrations thousands of times greater than that of the surrounding medium. If such algae entered the food chain of mobile animals, some contamination might be spread.

A recent Russian report[37] is concerned with the possible spread of contamination from radioactive-waste-holding basins by aquatic insects that emerge and fly away from the basin. The report gives the results of a small-scale laboratory experiment in which three kinds of common aquatic insects (caddis flies, May flies, and midges) were reared in aquaria which contained either Sr^{90}, Cs^{137}, Co^{60}, or concentrations of three-year-old mixed fission products. Depending upon the isotope and the concentration, the total activity per hatched insect ranged from 3×10^{-5} to 1.1×10^{-2} μc.

In all cases, when the individual insect emerged, it contained only a small fraction (0.0003 to 0.02 per cent) of the total activity in the aquarium. The report suggests that these levels of activity would be typical of insects emerging from radioactive-waste impoundments but that insects could cause "a comparatively rapid and complete bioecological self-purification" because of the large numbers of insects emerging and spreading out from confined basins.

The authors,[37] using the data of E. V. Borutskii, estimated that a total of 2×10^8 insects per year emerge from a typical basin (0.5 km^2) in arid mid-Russia. Of these, 1.8×10^8 would be eaten by predators; the remaining 2×10^7 insects would die and be deposited over an area assumed to have a radius of 1.5 km from the basin. The report suggests that, if this biological activity were continued over a period of time (2 to 10 years), assuming deposition of two insect bodies per square meter per year, the radioactivity in these nearby land areas would be twice that due to a single hydrogen bomb, which is given as 5.4 $m\mu c/m^2$.

The authors[37] appear to be aware, however, of some obvious weak points in this hypothesis. They note that the bottom muds and organic detritus will sorb the greater fraction of the radionuclide burden and thereby reduce its biological availability. Also, most of the activity absorbed by the larvae will return to the basin when the insect emerges (the larval skin remains behind). Furthermore, the time required to decontaminate an entire basin would exceed the half life of Sr^{90} because many of each generation of insects would be eaten by other organisms in the basin.

An almost complete lack of information on the numbers of individuals and population dynamics of aquatic insects makes it difficult to assess the applicability of these assumptions to other areas. The authors' basis of 5.4 $m\mu c/m^2$ of total radioactivity from one bomb is considerably less than the quantity already present from many test explosions,[38] which for Sr^{90} alone is estimated at 10 to 20 $m\mu c/m^2$. Consequently, the time estimates given by the Russians would have to be considerably increased to result in terrestrial contamination significantly above fallout levels. Other factors, such as blowing of contaminated dust, might become more important than the contamination from insects over a long time.

(S. I. Auerbach)

Retention and Movement of Radioisotopes in Soils

The use of soils for sorbing and retaining radioisotopes of wastes generated in the nuclear industry is a widespread practice. The reduction in mobility of isotopes sorbed by the ion-exchange complex of the soil results in an appreciable decrease in the hazards associated with the release of the radioactive wastes. The behavior in soil of the various radionuclides differs because of differences in the chemical nature of the isotopes and because of the mineralogical nature of the soils and sediments. In addition, the behavior of an isotope occurring as a microconstituent in an ion-exchange system is not necessarily the same as that of the same isotope occurring as a macroconstituent. The waste streams arising from various processes in the nuclear industry differ considerably in the chemical nature of their macrocomponents, especially salt content, as well as in the nature and concentration of the radioisotopes present. With these points in view, the necessity for defining the behavior of all c r i t i c a l radionuclides of the waste streams in soils of various mineralogical natures is clearly evident. Some of the recent findings regarding the behavior of individual radionuclides, as reported by investigators at several installations, are discussed here.

Plutonium

Several investigators[39–41] have found plutonium to be very strongly sorbed by soils. Rhodes[41] found that the uptake of plutonium was most favorable in the *p*H range from 2.2 to 8.4. He suggests that plutonium exists as a polymer in this *p*H range and that the plutonium sorption is nearly irreversible. Plutonium sorption was not found to be materially reduced by the presence of salts.[39,41] Christenson et al.[39] could determine no appreciable removal of plutonium from tuff when the leaching solution consisted of 100 ppm of Na^+, Ca^{++}, Mg^{++}, or saturated CO_2.

Cesium

The behavior of cesium in soils appears to be quite dependent on the nature of the soil minerals. Christenson et al.[39] found that cesium is strongly held by tuff from the Los

Alamos area with very little regard for other ions present in the water. Attempts to disoldge the sorbed cesium by leaching with solutions consisting of 100 ppm of Na^+, Ca^{++}, Mg^{++}, or H_2CO_3 were futile. In soils that were contaminated with Cs^{137} and Sr^{90}, it is also reported[39] that the Cs^{137} remained evenly distributed within the soil, whereas the Sr^{90} was found to leach from the soil and appear in the seepage from the bottom of 18-in.-deep plots.

Rhodes,[41] working with Hanford soils, found that the sorption of cesium could be lowered considerably by increasing the concentration of salts in the solutions. Rhodes also observed no measurable effect of *p*H on the sorption of cesium from distilled-water systems until the *p*H was lowered to 1.8 or less. This is probably due to the contribution of the H^+ to the total ionic activity of the solution, resulting in increased cationic competition with cesium for the exchange sites on the soil.

On the other hand, Prout[40] found the sorption of cesium to be *p*H dependent by the kaolinitic soils of the Savannah River area. He states that kaolinite is a weakly acidic clay and that, therefore, the exchange capacity varies with the *p*H of the system. Thus the *p*H seemingly affects the behavior of the clay rather than the behavior of the cesium.

The importance of the mica structure in sorbing cesium has been found for the Conasauga shale at Oak Ridge.[42] The presence of mica may explain the insensitivity of the Hanford soil, as compared with the kaolinitic Savannah River soil.

Strontium

There is general agreement[40,41,43] that strontium sorption is very sensitive to *p*H. Rhodes[41] and Prout[40] found that the nature of the cation used to adjust the *p*H was critical. Increasing the *p*H with NaOH improved strontium sorption until the competitive effect of the increased sodium ion concentration overshadowed the beneficial effect of increased *p*H. On the other hand, additions of $Ca(OH)_2$ caused a depression of strontium sorption at a lower *p*H than that with NaOH, indicating that the beneficial effect of increased *p*H was overcome by the competition of the calcium with strontium for the exchange sites. For the calcareous Hanford soil, a reduction in strontium sorption was noted at a *p*H of 10 for the NaOH system and at a *p*H of 9.2 for the $Ca(OH)_2$ system; for the acidic Savannah River soil, the decrease in strontium sorption was noted at a *p*H of 8 for the NaOH system and at a *p*H of less than 7 for $Ca(OH)_2$.

The favorable influence of PO_4^{-3} ions on the uptake of strontium by Hanford soil was first demonstrated by Rhodes.[41] At a *p*H of 8, for example, a distribution coefficient, K_d (see Table V-5 for the definition of K_d), of about 30 was found in distilled water; whereas in 4*M* NaCl the distribution coefficient was reduced to about 3. Addition of 0.01*M* PO_4^{-3} to the 4*M* NaCl solution increased the distribution coefficient to approximately 90. McHenry[43] showed that the oxalate ion also influenced the uptake of strontium in a favorable manner, whereas additions of NO_3^-, Cl^-, and SO_4^{--} were

Table V-5 MECHANICAL ANALYSIS, *p*H, TOTAL EXCHANGE CAPACITY, AND K_d OF SOILS

Soil	Location	Mechanical analysis, %			*p*H	Total exchange capacity, meq/100 g	K_d*
		Sand, >50-μ particles	Silt, 50- to 2-μ particles	Clay, <2-μ particles			
Ringold	Eastern Washington	18.6	39.0	42.4	7.2	34.0	400
Warden†	Eastern Washington	78.1	18.9	3.0	7.3	6.0	270
Bowdoin†	Eastern Montana	48.5	34.6	16.9	8.3	13.6	135
Hall†	Central Nebraska	5.9	60.6	33.5	6.5	26.3	600
Hanford Project composite		84.6	11.9	3.5	8.3	5.8	70

*Ratio of strontium per gram of soil to strontium per milliliter of solution.

†Exchange capacity, *p*H, and mechanical analysis supplied by Division of Soil Survey, U. S. Department of Agriculture, Mandan, N. D.

found to have an adverse effect on strontium sorption. Both Rhodes[41] and McHenry[43] considered that the anions influenced the ion-exchange behavior of the soil. Later studies at Hanford have considered the added factor of crystallization of calcium phosphate (apatite) in increasing the uptake of strontium.[44]

The study reported by McHenry[43] is concerned only with the factors affecting strontium uptake, and it is deserving of further comment. The competition for the exchange of strontium by the soil was found to increase with increasing ionic radii of the alkali metal cations. When Cs^+, Ba^{++}, and La^{+3} ions were compared in solutions containing $3 \times 10^{-6}M$ strontium or less, no effect of charge or concentration of the complementary cation was found. Effects of concentrations of the complementary cation were noted at strontium concentrations less than $3 \times 10^{-6}M$, although the data do not show conclusively the effect of the charge, as claimed in the paper.[43]

Although the statement is made that there is no apparent correlation between the distribution coefficient and the soil properties, there seems to be an inverse relation of *p*H and the distribution coefficient, as shown in Table V-5. Prout's data[40] showed that the distribution coefficient decreased when small amounts of $Ca(OH)_2$ were added to raise the *p*H of the acid soil. Thus, in comparing these soils, it may be concluded that, with increasing *p*H of the soil, there is the competitive influence of the calcium contributed by the hydrolysis of the limestone.

The *p*H of the Hanford composite soil is reported to be 8.3, and it can be assumed that an appreciable amount of calcium ion is in solution because of the dissolution and hydrolysis of the limestone. In this respect, it might be well to reconsider the desorption data reported by McHenry.[43] In addition to establishing the relative effectiveness series of $Li^+ < Na^+ < K^+ < Mg^{++} < Ca^{++} < Ba^{++} < La^{+3}$, the statement is made that leaching with water alone removed the adsorbed strontium. The effectiveness of the water was found to be approximately equal to $0.001M$ $NaNO_3$. Considering the dissolution and hydrolysis of limestone, the desorption of strontium from the soil may be due to the presence of dissolved calcium rather than to the innate ability of water for ionic desorption.

Ruthenium

Rhodes[41] found that the removal of Ru^{106} from distilled-water solutions was greatest with neutral solutions and that sorption decreased with increasing base or acid concentrations. At a *p*H of 1.3 and lower, there was no sorption of ruthenium. Prout[40] also noted no appreciable ruthenium sorption below a *p*H of 2. As the *p*H was increased to 5, Prout found a marked increase in the sorption of ruthenium. At a *p*H of 5, Prout reports that 2 to 3 per cent of the ruthenium could not be adsorbed, even when exposed to successive increments of fresh soil. At *p*H 6 and 8 the nonadsorbable form increased to 10 and 16 per cent of the total ruthenium. Lacy[45] found that, in solutions of *p*H 12, only 10 per cent of the ruthenium was in a form that could be sorbed by the Oak Ridge soil. Thus, on the basis of distribution coefficients, the expression of ruthenium sorption results may be misleading unless account is taken of the ruthenium that might exist in a nonadsorbable form.

Zirconium-Niobium

The sorption of zirconium and niobium from $1M$ HCl solution was found by Rhodes[41] to be negligible, but sorption increases rapidly as the acid concentration is reduced. He found that sorption was nearly complete between *p*H 4 and 7.7. At a *p*H of 9.6, a minimum is obtained; and, as the *p*H is increased to 12.2, sorption is again nearly complete. Column tests by Prout[40] at *p*H 4 and 8 also showed that zirconium and niobium were readily adsorbed by soil. The sorption from solutions at a *p*H of 4 was inhibited by $10^{-3}M$ oxalate ion, whereas the presence of oxalate did not seriously affect zirconium-niobium sorption from solutions of *p*H 8.

Other Radioisotopes

The behavior of yttrium and cerium was also reported by Rhodes.[41] In both cases the sorption was at a maximum from neutral solutions. This is not at all surprising since these solutions offer the least cationic competition for the exchange reaction. The data suggest that separation of a *p*H effect from a salt effect is extremely difficult in systems where addition of acid or base to effect a change in *p*H results in

an appreciable change in the ionic activity of the solution.

Although the papers reviewed here do not provide an intensive description of the behavior of all the important radionuclides present in the various nuclear waste streams, they do represent an initial attempt in the evaluation of the interactions between the waste streams and the geologic formations to which the wastes are disposed. As such, they provide valuable information and should provide a basis for a more intensive evaluation of the role of soils and sediments for the sorption and retention of radionuclides. (D. G. Jacobs and T. Tamura)

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NRU Reactor Incident

During the operation of the NRU reactor at Chalk River in May 1958, three fuel elements appeared to be releasing high levels of activity into the coolant stream. After a series of misadventures in removing the damaged fuel from the reactor, a large piece of highly active uranium was accidentally released from the fuel-handling equipment and fell to an exposed pit, where it burned and released large amounts of activity. As a result of the incident, the interior of the building was severely contaminated, and the reactor was shut down for several months. Despite radiation fields which were believed to be in excess of 1000 r/hr, the highest exposure to personnel during the incident was 19 r to one man. Only small amounts of contamination were released outside the reactor building, and detectable activity was confined to an area of about 100 acres.

A recent report by Greenwood[1] of Atomic Energy of Canada Ltd. briefly describes the events leading up to the activity release and then discusses in greater detail the problems faced and methods used in putting the reactor and its building back into operation. This review is concerned primarily with the events leading up to the incident and the causes thereof.

General Description of Reactor

The NRU reactor at Chalk River is basically an engineering and research reactor with a thermal power output of 200 Mw. The core consists of about 200 fuel assemblies which are moderated and cooled by heavy water. The fuel assembly is made up of five flat metal bars of natural uranium clad with 1S aluminum and contained in a 2.5-in. tube. The flat bars that were used in the initial loading of the NRU were fabricated by slipping the aluminum sheath over the uranium bar and then drawing through a die to press them into contact. Since there had been cladding failures of 13 of the original rods, the sheathing for replacement rods was applied by extruding a layer of aluminum onto the uranium bar as it moved through the die of an extrusion press. This process of sheathing the fuel resulted in a stronger bond between the aluminum and uranium, and therefore better fuel-element performance was anticipated.

Sixteen control rods are provided for control and safety of the reactor.[2] Since the reactor can be started up, brought to power, and maintained at a particular power level automatically, it is only necessary to set the controls to a predetermined final power and to prescribe a maximum allowable linear and logarithmic rate of change of power. The reactor is capable of being brought to power safely in 10 min or less at a logarithmic rate that can be set between 1 and 4 per cent per second and then at a linear rate between 0.2 and 2 Mw/sec. The reactor power levels off when the power is about 10 per cent below the demand value. Engineering considerations of thermal shock[2] require the rate of change of power to be limited to 2 Mw/sec.

In order to increase the safety of the reactor and, at the same time, to reduce the probability of shutdown, most of the components in the control system are provided in quadruplicate and are arranged in parallel channels.[3] If a signal in one channel disagrees with the other three signals, the channel is automatically disconnected. When two or more channels give signals, the reactor will shut down. Provision is also made for shutting down the reactor due to excess power (set at 20 per cent above demand power), excessive temperature of coolant, unusual flow conditions, or excessive radioactivity in coolant.

Operation of NRU Prior to Incident

In the six months of operation from the time the NRU went critical to the time of the incident, the NRU operated at various power levels up to 200 Mw. Greenwood[1] describes some of the problems encountered in this period, stating that most of the normal difficulties encountered in initial operations were overcome but that a few were still under investigation. Some of the prob-

lems encountered were (1) a high background of activity in the heavy water, caused by early cladding failures; (2) variations in power due to lateral movement of the control rods, caused by the coolant circulation (power changes as high as 2 Mw/sec were frequently observed); (3) the system for monitoring flow in individual fuel rods did not operate satisfactorily; (4) the initial method of applying the sheathing to the fuel bars was not satisfactory, and, as mentioned above, a new process was being used for the replacement rods; and (5) the system for automatically correcting reactor power readings in terms of exit cooling temperatures was undergoing adjustment and was not connected to the control system. The absence of this latter system did not influence the control system, however, since power readings were taken directly from the ion chambers.

Operation of NRU at Time of Incident

Greenwood[1] describes the events at the time of the incident as follows:

> On May 23, 1958, after a week of steady operation, the reactor suffered an automatic shutdown as a result of excessive power rate-of-rise, but the staff could find no evidence to account for the occurrence. Therefore they started the reactor again, only to be met with another automatic shutdown almost immediately. This time the excessive power rate-of-rise was accompanied by alarm signals indicating a number of unusual conditions in the reactor, the most significant of which was very high radioactivity in the coolant circuit. Some of the other signals were later shown to have resulted from a pressure transient inside the reactor vessel arising from violent failure of one of the fuel rods.

The behavior of the control system and the cause of the events of May 23, as described by Greenwood, may be summarized as follows:

1. The two excess-linear-rate trips were recorded on all four channels and were therefore believed to be genuine and probably due to a rapid change in neutron flux at the ion chambers which was too fast for the recorder to follow. (The first trip occurred during supposedly steady operation, and the final one occurred while reactor power was apparently rising at 1 Mw/sec; both were well below the trip level of 5 Mw/sec.)

2. The phenomena observed at the time of the final trip can be explained by postulating a pressure transient within the reactor vessel due to the bursting of a fuel rod. A lowering of the level of the heavy water could have generated a withdrawal signal to the control rods even though they were operating perfectly. The reason for the first trip is unknown.

3. After intensive studies of the control system, it was possible to explain the power overshoot with near certainty by a weak spring on the "low power–normal" selector switch which was used to initiate the automatic start-up. If this switch were not fully turned from "low power" to "normal," some of the necessary electrical contacts would not be made, and therefore the reactor power would behave in exactly the anomalous fashion that was observed.

It is possible, therefore, that during normal operation prior to the first shutdown, a fuel rod developed a minor defect as the power drifted up to 206 Mw(t). Subsequent attempts to bring the reactor to power increased the degree of "waterlogging" in the fuel so that, prior to the final trip, a condition existed for a large pressure transient in the core. If the power was rising at a rate of 3 per cent per second instead of 1 Mw/sec because of the failure of the selector switch, when the pressure transient occurred, an overshoot in power was possible which would result in the violent bursting of the fuel rod.

Although the failure of fuel elements is to be expected, especially during initial operation with a new type of fuel element, the fact that the original fuel rods for the NRU had been unsatisfactory would make it even more desirable to have a sensitive monitoring system to detect the failure of a rod before the failure became progressively worse. At the time of the first trip, the monitoring system was not capable of detecting a minor rupture, and therefore the fuel was subjected to power transients that led to a severe rupture.

The rapid rise in power due to the weak spring in the automatic start-up system represents another weak point in the control system. Adequate circuits to ensure that the proper contact has been made could prevent a rapid rise in power. (Greenwood[1] reports that such changes have been made in the circuitry.)

Activity Release During Removal of Ruptured Fuel Elements

At the time of the final trip, large amounts of radioactivity were detected both in the coolant

and in the building. Subsequent examination revealed that three fuel elements had possibly been damaged.

The normal fuel-removal sequence, as described by Greenwood,[1] is set forth below. The fuel-removal flask is moved into position over the appropriate rod and is sealed to the uncapped hole by a metal snout on the end of an extendable bellows. The empty chamber of the flask is then filled with heavy water. The "main valve" at the bottom of the flask chamber is opened to create a continuous passage from the reactor to the flask, and the guide tube is driven down from the upper shielding section of the reactor to the fuel rod in the core. The rod assembly is then raised until the fuel is entirely within the upper shielding section, where it remains for about 15 min until some radioactivity has decayed; during this time it is being cooled by water rising through the guide tube. The fuel assembly is then raised completely into the flask, where it is cooled by heavy water from the built-in circulation system. Another chamber in the flask, which was previously loaded with a fresh fuel rod, is then rotated into position over the open hole. The new fuel assembly is lowered through the guide tube into the reactor. The guide tube is then raised to its normal position in the upper shielding, and the main valve is closed. The bellows seal between the flask and the deck plate is broken; the flask shielding skirt is raised; and the flask is moved to the storage block. The dust cover is replaced in the reactor hole.

Although one of the fuel rods thought to be damaged was removed with no difficulty, the second damaged fuel rod was removed with near-disastrous consequences. Some of the causes of the difficulty in removing the second fuel rod, as described by Greenwood,[1] are summarized here.

As a result of probable damage to the fuel rod, the guide tube could not be lowered completely around the fuel in the reactor. Since the damaged rod could not pass through the guide tube, it was necessary to remove the entire tube-mechanism (TM) plug. This required that the lower section of the removal flask be altered to accommodate the TM plug. When the fuel and TM plug were raised into the upper shielding of the reactor, it was discovered that the heavy water had drained out of the flask chamber into the reactor, leaving no means of cooling the fuel once it reached the flask. During normal removal this is prevented by the upward pressure of heavy water from the guide tube. Although the operating crew were aware that without the guide tube in place the pressure would be much less, they expected the pressure to be sufficient to raise the heavy water to the flask pump intake. (Greenwood[1] points out that it is now known that the static pressure without the guide tube will raise water to a point 2 or 3 ft short of the pump inlet.) To assure that the absence of cooling water was not due to blockage of the passage by the TM plug, the operators raised the fuel and TM plug above the main valve; but, when there was still no sign of rising heavy water, an unsuccessful attempt was made to return the fuel to the reactor. Although several actions were taken to replenish the flask with heavy water by other means, none of the attempts was successful because of lack of time and because the flask pumps had gas-locked from loss of water.

After it was determined that the fuel was within the flask chambers and that the flask circulation system would definitely not operate, an attempt was made to move the flask to a point just off the reactor deck where a hose was ready to supply ordinary water for emergency cooling. On the first attempt the drive motor stopped immediately since the snout had dropped away from the retracted position and caused the flask drive motor to stop through an electrical interlock. At this time it was noticed that the main valve had opened, and it was thought that the end of a fuel rod was hanging out of the snout. Although the crew thought that the valve was blocked by the fuel rod, it was later established that there was a malfunctioning of the hydraulic control valve. Attempts to close the valve were unsuccessful because an electrical interlock prevented the valve from operating unless the snout was fully down.

Since the snout could not be raised, the snout interlock was jumpered to permit the flask to be moved in order to clear the reactor deck, and emergency cooling water was attached to the flask as soon as possible. Because of the faulty main valve in the bottom of the flask, contaminated water leaked out as the flask traveled to the storage block. It was during this time that a piece of fuel dropped into the maintenance pit.

In discussing the causes of the incident, Greenwood[1] draws several important conclusions of interest to both operators and designers

of nuclear facilities. Among these he states that:

> A highly active element was transferred to a less safe position without first following the standard policy of ensuring that all equipment was operating properly. This points out the ever-present dilemma facing those in charge of a complex operation such as the NRU reactor: whether to leave some discretionary power in the hands of the operating crew and accept the increased chance of an accident or to operate by strict procedure and accept the resulting disadvantages, of which one of the most undesirable is the inferior opportunity for training staff and for retaining their interest in work.

He further states that:

> An incidental observation arising out of the fuel removal is the surprising fact that the presence of certain "safety" interlocks actually increased the hazard by preventing actions that, although not normally desirable, needed to be carried out urgently under the emergency conditions at the moment. One might conclude that reactor designers should pay special attention to the function and detailed mode of operation of all safety interlocks, making sure that, as far as possible, the restrictions they create are necessary and desirable under all conditions.

It is the opinion of the reviewer that, in regard to operation, strict adherence to operating procedure, with consultation with the most experienced and responsible supervisors, is the correct path to follow, unless the emergency condition requires that immediate action be taken. Training programs, procedures, and special equipment for all foreseeable emergency conditions must be used to help new personnel cope with foreseeable emergencies. Furthermore, the incident points out the importance of detailed studies preceding various operations. Although the operating crew at the time of the incident may have been imprudent in removing fuel to a less safe position, it is even more imprudent to provide an operator with a facility that will not cope with probable malfunctioning. Since there was a distinct possibility that the guide tube would not always be in a position to provide coolant to the flask, it is apparent that emergency heavy water should have been available to the flask at all times and that a reliable system of indicating water flow and temperature inside the flask was needed. Furthermore, although interlocks were provided for restricting operation of the flask under normal circumstances, a detailed study should have been made to ascertain the consequences of these interlocks at the time of emergency. The valve in the flask (which did not operate because of a malfunction of the hydraulic control valve) could, if it were fail-safe designed, prevent the spread of contaminated water. It appears that greater emphasis was placed on preventing contamination of the heavy water in the reactor than in preventing the spread of activity.

Considering the potential damage possible in this incident, the Chalk River staff did a remarkable job in recovering without severe injury to personnel. They have quite frankly presented their experience, from which the nuclear industry can learn and profit. (A. F. Rupp)

Sodium Reactor Experiment Incident

On July 24, 1959, the Sodium Reactor Experiment (SRE) was shut down to investigate abnormalities which prevailed in the operations during power run 14. A subsequent preliminary examination revealed that extensive damage had been sustained by several fuel-element clusters during this power run. An internal *ad hoc* committee was convened by Atomics International, operator of the reactor, to investigate the failure. On Aug. 14, 1959, the General Manager of the AEC directed, pursuant to a recommendation by the Division of Reactor Development, an Investigation Review Committee to review the circumstances and the field investigation thereof relative to the abnormal operations and fuel-element failures which occurred at the SRE. Accordingly, the Chicago Operations Office of the AEC, having jurisdiction over the reactor, appointed a committee consisting of personnel experienced in sodium technology from AEC and affiliated installations to review the reactor operations immediately preceding the incident and to make recommendations for rehabilitating the reactor system.

Two recent reports by Atomics International[4,5] describe the facts pertinent to the SRE incident, and some of the preliminary conclusions are discussed in this Review.

The SRE reactor at Santa Susana, Calif., was built to develop this reactor concept for civilian power application. The reactor is moderated by graphite, cooled with sodium, and contains 43 channels for fuel elements in the center of the moderator cans. The design outlet temperature

of the sodium is 960°F. The graphite is canned with zirconium to prevent sodium from entering the void spaces of the graphite. The fuel elements in the SRE are fabricated in clusters of seven rods, each rod consisting of a 6-ft-high column of uranium slugs in a thin-walled stainless-steel tube. A thermal bond between the uranium and the stainless-steel jacket is obtained by using NaK in the annulus, and helium is contained in a space above the NaK to allow expansion of the NaK and to serve as a reservoir for fission gases escaping from the uranium. A more detailed description of the reactor is given in references 6 and 7.

The Atomics International *ad hoc* committee report describes the reactor operation for the period from Nov. 29, 1958 (beginning of run 8), to July 24, 1959 (end of run 14), and provides background information necessary to an understanding of the present condition of the reactor and how it came about. Some of the more important events during this period are discussed below.

During run 8 a large spread existed in the fuel-channel exit temperatures; this spread was attributed to the high oxygen content in the sodium which caused oxide plugging in the process tubes. Action was taken to reduce the oxygen content, and the fuel (which had been running excessively hot) was removed, washed, and then returned to the reactor. During run 8 a sample of the core cover gas indicated that Tetralin, an organic compound used as an auxiliary coolant, had entered the primary system.

During runs 9 to 11 the fuel-channel exit temperatures continued to show large spreads but tended to improve as the various runs proceeded. At the end of run 11, fission-product contamination was detected in the sodium, but it was not excessive. Run 12 exhibited no abnormal behavior. During this run the sodium outlet temperature was raised to 1065°F for a short period, and steam was produced at 1000°F.

In run 13, which was a high-temperature run with a 1000°F sodium outlet temperature, after an initial scram as a result of an abnormal sodium flow rate, the reactor was returned to normal operating conditions. Several unusual situations then arose: the reactor inlet temperature started a slow rise; the log mean temperature difference across the intermediate heat exchanger started to increase, indicating changes in the heat-transfer characteristics; a thermocouple in a fuel slug in channel 67 showed an increase from 860 to 945°F; some of the fuel-channel exit temperatures showed slight increases; and the temperature difference across the moderator abruptly jumped 30°F. Later examination indicated that a reactivity increase of about 0.3 per cent occurred over a period of about 6 hr and then increased about 0.1 per cent over the next three days of operation. By June 2, 1959, it was obvious that the heat-transfer characteristics of the system had been impaired. Since the sodium oxygen content had been low at the start of the run, it was decided that a Tetralin leak was causing the trouble instead of excessive oxygen in the sodium. Subsequently, Tetralin was detected in the pump casing of the main primary pump; and, after the run was terminated, a leak was found in the thermocouple well in the freeze seal of the pump. Examination of the fuel by television camera indicated the fuel to be "slightly dirty but in good condition."

On June 4, 1959, while the fuel element from core channel 56 was being washed, a pressure excursion occurred which severed the fuel hanger rod and lifted the shield plug out of the wash cell. It is postulated that the hydrocarbons (from the Tetralin leakage) could cause sodium to be trapped in the hold-down tube on the hanger rod by blocking the sodium drain holes, and the sodium could cause the reaction. As a result of the "wash-cell incident," no further washing of elements was done. Prior to run 14 the Tetralin-cooled freeze seal in the sodium pump was replaced by a NaK-cooled unit, and Tetralin and naphthalene, a product of Tetralin decomposition, were removed from the primary system.

During the initial phases of run 14, some of the abnormal conditions noted in run 13 were again observed (i.e., large fluctuations in the temperature difference across the moderator and divergence of the fuel-channel exit temperature); however, reactor operation continued at low power (1 Mw) until a scram was caused by loss of auxiliary primary sodium flow. Shortly after reestablishing operation, a sharp increase in air activity was indicated in the reactor room, followed later by a sharp increase in the stack activity. Since the radiation level over channel 7 had reached 25 r/hr, the reactor was gradually shut down. The activity escape was traced to the sodium-level coil thimble which was replaced by a shield plug. After returning the re-

actor to operation, no significant activity was noted in the high-bay area.

On July 13, 1959, a series of negative and positive reactivity excursions was observed; one of these excursions resulted in a 7.5-sec period. The reactor was scrammed manually. It is estimated that the reactor reached a peak power of 24 Mw(t). The cause of the reactivity changes is not known, but investigations are being made in an attempt to explain them.

From the preliminary findings of the Atomics International committee, it may be concluded that:

1. The fuel-element failures resulted indirectly from leakage of Tetralin into the primary sodium circuit. The mechanism of failure is thought to have been either the blockage of coolant passages or the fouling of fuel elements by the products of Tetralin decomposition, which caused subsequent overheating of some fuel elements.

2. The fuel-element temperatures rose sufficiently to induce eutectic melting between the uranium and the iron in the type 304 stainless-steel fuel cladding.

3. Complete melting of the cladding around 10 of the 43 fuel assemblies in the reactor is now known to have occurred. The resultant loss of cladding support led to a complete separation of the top and bottom halves of these 10 assemblies. In every case the zone of fracture was between one-third and two-thirds of the length measured from the top of the elements.

4. The iodine released at the time of the fuel-cladding failure appears to have been retained very effectively by the sodium coolant.

5. The reactor excursion during run 14 is attributed to the rapid addition of reactivity and to the failure of the setback circuit to initiate proper corrective action. This circuit is actuated mechanically by means of a cam in the reactor period recorder; a notch in the cam trips a switch that initiates reduction of reactor power by proper rod action. Although the circuit was set to operate when the period reached 10 sec, examination of the mechanism indicated that it operated satisfactorily only if the period decreased at a fairly slow rate and that, when the period decreased rapidly, the switch would fail to open. Modifications have been made to assure satisfactory operation at all rates of change of the reactor period.

6. The difficulties encountered at the SRE are not attributed to the use of sodium as a coolant but rather to the impurities that were introduced into the coolant.

At various times prior to run 14, there were indications of difficulties arising from Tetralin leakage into the system, as well as fission-product leakage into the coolant. It is indicated that, following the wash-cell incident, fuel elements were not washed; therefore it is probable that the fuel in the reactor contained substantial amounts of Tetralin, which may have caused the final difficulty. It is the opinion of the reviewers that the reactor instrumentation under the immediate surveillance of the operator was inadequate to indicate excessive fuel-element temperatures, the blocking of coolant passages, and fission-product leakage. As a result the operators did not consider such indications (where they existed) serious enough to warrant shutting down the reactor. Since the SRE is a "developmental facility built to investigate fuel materials," it would appear that additional instrumentation, as well as closer technical management, might have reduced the damage to the SRE core.

(W. B. McDonald and J. H. DeVan)

Idaho Chemical Processing Plant Criticality Incident

On Oct. 16, 1959, at 2:50 a.m., a nuclear incident occurred at the Idaho Chemical Processing Plant (ICPP), a shielded facility which recovers uranium from highly enriched irradiated fuel elements. Seven of 21 persons directly involved in the incident received significant external radiation exposure, with a maximum individual exposure of 50 rem. The incident has been described in the report of an investigating committee.[8] The incident was caused by the accidental transfer of approximately 34 kg of highly enriched uranium (91 per cent U^{235}) in aqueous solution from 5-in.-diameter pipes into a 5000-gal vessel. The excursion resulted in about 4×10^{19} fissions before shutting itself down by blowing solution and/or vapor into a partly filled vessel of negligible uranium content.

At the time of the incident, the plant was processing stainless-steel fuels. Since the capacity in the initial recovery steps (dissolution and first extraction cycle) for this type of fuel is lower than in the balance of the process (sec-

ond and third extraction cycles), uranium is accumulated and stored before the final operations are performed. The solution was stored in two banks at a concentration of approximately 170 g of uranium per liter of solution. Each bank consists of eight interconnected 5-in.-diameter 10-ft-high pipes. The two banks have a common drain line which is provided with a jet suction line to a waste header for transferring decontamination solutions. Prior to the incident each of the banks contained approximately 42 kg of uranium.

It was decided to sample the banks since they were almost 80 per cent full. Prior to and during sampling it was standard procedure to agitate the solution with air spargers. Each vessel in the storage bank has a sparger consisting of a $\frac{1}{2}$-in. pipe with the lower end plugged and two $\frac{1}{8}$-in.-diameter holes drilled through the pipe near the bottom of the vessel. The air pressure upstream of the sparger inlet valve to each bank is 50 psig. Because of limited use, flow-restricting orifices had not yet been installed on these sparger supply lines, as had been done elsewhere in the plant. The sparger supply valve to one of the banks was opened until 2 psig was indicated. The sparger supply to the other bank was opened, but the pressure was not read because the panel pressure gauge was inoperative. The operator did not know that another gauge, not on the panel, had been installed on the line. Therefore the operator closed the valve and reopened it until the movement of the pen recording the density of solution in the bank indicated that sparging was occurring.

The liquid-level and density instrument charts indicate that, beginning shortly after the sparging was started, the solution siphoned for about 15 min from the first bank via a drain line to a 5000-gal waste-storage tank. The latest information[9] indicates that the transfer was probably initiated by excessive sparging of the geometrically safe banks, and pressurization of the banks resulted. This pressurization was apparently sufficient to force the uranium solution up and out the jet drain line, the highest point of which was 4 ft above the maximum liquid level in the banks. The flow apparently continued, owing to the siphoning action, after it was initiated by the pressurization.

Approximately 200 liters of solution containing about 170 g of the 91 per cent assay uranium per liter flowed from the storage banks through a diversion spout to a waste tank that contained about 600 liters of aqueous solution of negligible uranium content. The diversion spout had been set to this almost empty receiver prior to the incident because its companion vessel was almost full. Both 5000-gal vessels are 50 ft below grade and are covered with a 4-ft-thick concrete deck.

The nuclear excursion occurred in the almost empty receiver. The system went critical and returned to subcritical under conditions unknown to, and unsuspected by, operating personnel. The actual mechanism or duration of the excursion is not determinable, but it is known that the reaction was shut down by the expulsion of 600 liters of solution (of the 800 liters then in the vessel) presumably into the almost full waste receiver. The vessels did not rupture, but the diversion spout was forced back into a position that caused it to drain into the almost full receiver. Subsequent Mo^{99} analysis of the solution revealed that approximately 4×10^{19} fissions occurred before the reaction was terminated. From Fe^{59} and Co^{60} activation of stainless-steel pieces near the vessel, an integrated flux of about 1.5×10^{13} neutrons/cm^2 was estimated.

The first indication that there was an abnormality in the process building came as a result of the spread of gaseous and air-borne beta and gamma contamination from the waste tank area, possibly through vent lines and drain connections, to areas where radiation monitors were located. Alarms actuated by these monitors resulted in the evacuation of the building. Contamination was also spread outside the building by air-borne particulate and/or gaseous beta and gamma activity which entered the unfiltered vessel off-gas system and was discharged through the local plant stack. At the time of the building evacuation, the radiation field outside the building and for 130 yd to one side of it was 5 r/hr or greater. Forty-five minutes after the evacuation, the air-borne activity had dissipated sufficiently for personnel to reenter the building and shut down the equipment.

Absence of blood sodium activation indicated that there were no personnel neutron exposures. Since iodine release from another operation was initially suspected, potassium iodide was administered orally. The film-badge dosimeter data and internal exposures calculated for the three people who received the largest radiation doses are given in Table VI-1.

Table VI-1 FILM-BADGE DOSIMETER DATA AND INTERNAL EXPOSURES CALCULATED FOR THREE PERSONS EXPOSED AT ICPP

	Employee		
	A	*B*	*C*
External beta exposure, rem	44.2	24.2	6.45
External gamma exposure, rem	6.0	8.0	3.95
Total external exposure, rem	50.2	32.2	10.4
Sr^{89} dose to bone, mrem	9	2	15
Sr^{91} dose to bone, mrem	3.5	Negligible	3.9
Bone dose from internal emitters, mrem	13	2	19
Sr^{89} dose to GI tract, mrem	4	Negligible	5.9
Sr^{91} dose to GI tract, mrem	23	14	23
GI tract dose from internal emitters, mrem	27	14	29

In the preliminary report on the incident,[8] several points were not completely resolved. These include the mechanism of flow initiation and stoppage from the banks to the waste vessel, the duration of criticality, the path taken by the contaminating activity from the critical vessel to the operating areas of the building and the building environs, the mechanism of reversal of the drain spout during the incident, and the absence of any inhalation of activity. It is anticipated that some of these questions will be resolved in a more detailed report which will follow. The investigating committee recommended (1) intense analysis of equipment and procedures prior to initial use or reactivation of the plant after modification or long downtime; (2) several lines of defense or at least a warning against criticality occurrence; (3) reevaluation of radiation and evacuation alarm systems and emergency procedures; and (4) consideration of placement of neutron detectors throughout the plant, even in areas where the probability of nuclear incidents is low.

This accident has served to reemphasize the necessity for very careful analysis of the transition between critically safe constructed equipment and other, not inherently critically safe, equipment connected to it. Such a transition is required in almost all plants that handle fissile material. In addition to the provision of sufficient interlocks and procedures to prevent the direct activation of a transfer device by operator error, consideration must be given to one or more successive abnormal equipment failures or operations that could result in an inadvertent transfer of fissile material. Often it is possible to minimize the possibility of pressurization of a vessel by providing flow restrictors in sparger lines (as is done at the ICPP) and extra large vent lines. In instances where a jet must be connected from a "safe" vessel to a "nonsafe" vessel, it is often possible to minimize the possibility of an inadvertent siphoning action by (1) mounting the jet relatively high above the "safe" vessel and (2) providing a valve that automatically vents the jet when the motive air or steam flow is cut off.

This incident has also indicated that the seriousness of such an accident would be considerably lessened by the provision of an adequate vessel and cell ventilation system. Such a system, if properly constructed and maintained, could significantly limit the spread of activity to the environment. "Absolute" filters for such ventilation systems should be considered almost mandatory, and serious consideration should be given to the provision of a unit for removing the relatively volatile radioisotopes of iodine and ruthenium which are formed in a nuclear excursion. (J. W. Ullmann and J. P. Nichols)

Activity Releases at Oak Ridge National Laboratory

During the period between October 28 and November 20, 1959, three incidents involving the accidental release of activity occurred at ORNL. The first release of activity occurred during the period between October 28 and November 6 when high-level radioactive waste was released into the process waste system accompanied by a discharge of approximately 55 curies of Ru^{106} and Ce^{144} into White Oak Creek. The second uncontrolled release of activity occurred on November 11 and 12, at which time about 15 curies of Ru^{106} was released from the gaseous waste disposal system through the brick stack at X-10. The third of these incidents involved an explosion in the Thorex Plant at X-10 releasing 1 to 5 g of plutonium into the building cell.

The plutonium release from the Thorex Pilot Plant is discussed in a separate article in this issue. The other two incidents will be reviewed in the next issue of *Nuclear Safety*. That issue will also include a discussion of safety in the light of these incidents and the action taken by ORNL to minimize the chances of such releases in the future.

Plutonium Release from the Thorex Pilot Plant

An explosion occurred in Cell 6 of the ORNL Thorex Pilot Plant on Nov. 20, 1959. Significant amounts of plutonium were released throughout the Pilot Plant (Building 3019) and the Graphite Reactor Building, and nearby streets and exterior building walls were contaminated. A recent report (as yet unpublished) by the ORNL investigating committee (C. E. Winters, Chairman; K. A. Kraus; and A. C. Upton) describes the events leading up to the accident and makes several recommendations to prevent the recurrence of such an accident. At the time of the writing of the committee report, a detailed examination of the cell and equipment had not been made because the high radiation level prevented access to the cell. There was no significant personnel exposure as a consequence of the accident, although it has been estimated that from 1 to 5 g of plutonium was released within the cell, and that of this amount 0.6 g escaped outside the building. This review, which is primarily a condensation of the original report, should be considered preliminary.

Description of Equipment and Events Leading Up to the Accident

Strong circumstantial evidence indicates that the explosion on November 20 occurred in an evaporator where the oxidizing agent was nitric acid and the reductant was either the decontamination agent (consisting of a then unknown mixture of alkaline salts, amines, hydroxy acids, phenol, surface active agents, and water) or possibly a collection of TBP (tributyl phosphate) and diluent solvent plus a substantial portion of radiation-degradation products of TBP and the solvent.

The "evaporator" is not a single piece of equipment. It consists of a vertical steam-heated section of critically safe design, a stripping column, a vapor separation tank, and connecting piping as shown in Fig. 5. Steam and liquid mixtures from the vertical steam-heated section rise into the separator, and the liquid returns to the bottom of the evaporator for another pass. Vapors from the separator pass upward through a packed column in the stripper and on to a condenser and storage tank. Feed for the evaporator is usually the aqueous phase

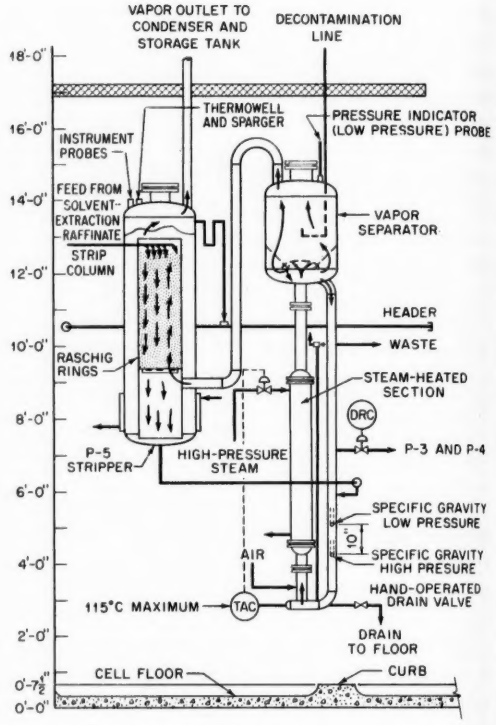

Figure 5—P-15 evaporator.

stripped from the first solvent-extraction cycle. In the current operations it contained uranium and plutonium. It also contained small quantities of dissolved solvent and solvent-degradation products. The purpose of this stripping operation is to steam-strip or distill organic material to prevent it from entering the evaporator. The degradation products are known to be less volatile and insoluble under these conditions and are presumed to collect on the packing in the stripper. Both mono- and dibutyl phosphate are known to form insoluble and nonvolatile metallic salts with metals, including plutonium.

Two days prior to the accident, the decontaminant had been added directly to the evaporator section and heated. It was followed by a water wash and then by a 30 per cent HNO_3 treatment. The decontamination had not yielded the desired results, and therefore another, more vigorous attempt was made. Accordingly, 200 liters of the decontaminant was added to the condensate tank and jetted to the steam stripper, which drains into the evaporator. After boiling for 2 hr, the decontaminant was drained. It should be pointed out that the normal drain line

enters the evaporator above the lowest point and that it was impossible to drain the last 15 liters except through a hand-operated valve. This valve can be operated only by entering the cell, which was not done because of the high radiation background. Two-hundred-seventy liters of 20 per cent HNO_3 was then added directly to the evaporator and boiled for about 2 hr, resulting in the concentration of the HNO_3. The remotely operated evaporator drain valve was opened; and, while the draining was in progress, an explosion occurred. About 200 liters of approximately 1.5*M* HNO_3 was found later in the condensate tank, and it provided a basis for calculating the residual HNO_3 concentration in the evaporator, i.e., 60 to 68 per cent HNO_3.

It is to be noted that the decontamination operation, during which the explosion occurred, differed from the normal procedures in the following respects:

1. The evaporator was not water washed between the addition of the decontaminant and the acid, although water washing was specified in the recommended procedure when using such materials and had been done in all previous instances. As a result of the procedure which was used, about 15 liters of the decontaminant remained in the evaporator when the acid was added.

2. The stripper had not previously been washed and boiled with the decontaminant, and an unknown amount of either phenol from the decontaminant or other organic residue was probably loosened and flushed into the evaporator as it was being drained.

3. The volume of nitric acid plus dilution water from jetting plus the calculated residual decontaminant in the evaporator was sufficient to flood part of the packing in the steam stripper. The liquid level remained up to the packing in the steam stripper during the following boiling cycle.

4. When the HNO_3 was boiled and evaporated, not only was the acid concentrated but also the recommended evaporator temperature based upon previous experience with similar materials was exceeded. In previous operations the evaporator had been maintained at a temperature below the nitric acid boiling point.

Conclusions and Recommendations

In summing up the causes of the incident, it may be seen that a dangerous procedure had evolved as a consequence of a number of procedural changes. Furthermore, the evaporator could not be drained completely. The recommended procedures for using the decontaminant, including the water rinse, were omitted. The reagents were boiled vigorously since heating almost always improves decontamination efficiency, and it was thought that the steam might help clean out the stripper. The steam was left on the evaporator during the draining operation to prevent precipitation that might have occurred during cooling. This procedure provided not only the easily nitrated organic compounds (such as phenol) and concentrated nitric acid but also provided the required heat to initiate a violent reaction. Taken singly, any one of the short-cut "operational improvements" might have been satisfactory, but in the aggregate they were not.

A serious error of judgment occurred on the part of the plant operators both in using substances whose composition and hazards were unknown to the operators and using them under conditions such that the substances came in contact with concentrated nitric acid. This error might have been avoided by stricter administrative control. The use of strong reagents, especially strongly oxidizing agents, including HNO_3, $KMnO_4$, etc., or reducing agents in any facility must be critically evaluated. If such reagents are required, the facilities employed must be designed for the added hazards.

The facilities in use were inadequate to withstand minor explosions without subsequent release of radioactivity to the environment, and the consequences of this accident could easily have been much more severe. Fortunately, the accident occurred when few people were in the vicinity; the weather was generally favorable; most of the radioactivity had already been removed from the equipment by prior decontamination operations; and, most important, prompt and effective measures were taken by all concerned to prevent further spread of contamination.

The lack of reliable alpha-survey meters, comparable to existing beta-gamma survey meters, hampered cleanup operations in that there was some delay in the correct identification of the problem. Unfortunately, the state of the art with respect to alpha monitoring leaves much to be desired. Alpha activity is generally determined by laboratory analysis of "smear" samples.

Many of the recent incidents in the United States have occurred during nonstandard or nonroutine operations. This implies that the same degree of planning and careful execution is not always applied during such operations, which, coincidentally, may in fact require the highest degree of administrative control. It must be recognized, however, that administrative control alone cannot prevent all accidents.

The containment requirement for a radiochemical facility is generally determined by the release of energy from the worst credible chemical explosion. The facility in point did not meet this requirement. ORNL has shut down all high-level radiochemical operations until the hazards are re-evaluated and appropriate corrective measures are taken where necessary. Not only is ORNL carefully scrutinizing the containment problem, but also decontamination and cleanup procedures — which should be considered an integral part of processing operations. The authors believe that a fresh look at the hazards associated with radiochemical operations generally is in order, especially in older facilities wherein the containment provisions may not have kept pace with changes in operations therein.

(H. N. Culver and W. B. Cottrell)

Activities of the Advisory Committee on Reactor Safeguards

At its meetings in October and November 1959, the Advisory Committee on Reactor Safeguards (ACRS) heard presentations involving eight reactors and sent letters to the AEC approving, with restrictions, amendments to the operation of two of the facilities. The two applications which were approved and which are discussed below were for (1) an increase in the operating power level of the Westinghouse Testing Reactor from 20 to 60 Mw(t) and (2) an increase in the operating power level of the Experimental Boiling Water Reactor from 20 to 100 Mw(t). Although comparable approval is not required for Department of Defense reactors, the ACRS also submitted its comments on the Heat Transfer Reactor Experiment No. 3A, which is also discussed below. Other applications which were discussed, but which were not approved either because the presentations were only informative in nature or because the information presented was inadequate, included Humboldt Bay Project (construction permit), Yankee Atomic Electric Company (operating license), Hanford New Production Reactor, and Dresden Nuclear Power Station (the hearings are discussed in a later section of this Review).

Westinghouse Testing Reactor (WTR)

The WTR[10] is a heterogeneous natural-water-moderated and -cooled low-pressure and -temperature general testing reactor that was originally approved for operation at a thermal power level of 20 Mw with an average thermal neutron flux of 5.2×10^{13} neutrons/(cm^2)(sec). In many respects the WTR is similar to the MTR; in addition, it is housed in a steel containment vessel. In support of the application to increase the thermal power level to 60 Mw, Westinghouse Electric Corporation had performed studies[11,12] which indicated that the hot-channel factors, film rise, thermal and hydraulic characteristics, and other pertinent features compared favorably with similar features of the MTR and ETR. In addition, the WTR has demonstrated the effect of bubble formation on reactor power-level fluctuations with the use of special bubble-generating equipment installed in a special channel isolated from the rest of the core. After examining the application and supporting data, the ACRS concluded that, based on the following considerations, the reactor could be safely operated at 60 Mw(t): (1) the special channel to determine the effect of bubbles should be retained until the 60-Mw level is reached, (2) the heat flux should not be allowed to go above one-half that required for burnout, and (3) no more than 1 per cent of the core volume should be voided by boiling.

Experimental Boiling Water Reactor (EBWR)

The EBWR[13] is a heterogeneous natural-water-moderated and -cooled high-pressure and -temperature experimental reactor that was originally approved for operation at a thermal power level of 20 Mw. The principal experimental feature of the plant is the natural-convection circulation of the water coolant as it is allowed to boil in the reactor core and to circulate directly to a steam turbine. The specific purposes of the proposed increase in power to 100 Mw(t) are to determine fundamental factors that would limit the ultimate power density of this type of reactor and to obtain operating ex-

perience at high power. A means for predicting instability before the power density at which the instability occurs is actually reached has been developed, and a high degree of leaktightness in the containment system has been demonstrated. After studying the application and supporting data, the ACRS concluded that, with the following restrictions, the reactor could be safely operated at 100 Mw(t): (1) the shutoff valves to the external heat exchanger should be of fail-safe design, be closed by low primary system pressure, or be closed by radioactivity in the stack, as well as by normal interlocks; (2) steps of power increase at which studies are to be made should be such that predictions could be made based on existing measurements that would demonstrate that the contemplated power increase would result in stable operation; (3) at least one period instrument in the scram system should be interlocked so that it would be active until the power coefficient noticeably reduced reactor power; and (4) this reactor installation should be given a rigorous periodic inspection equivalent to that given to commercial power reactors by the AEC Division of Inspection.

Several aspects of these restrictions are unique. The ACRS specifically commented upon the experimental nature of the facility and the need for safety information on instabilities. The location of the facility and the competence of the EBWR staff were cited as reasons to permit ANL to "venture cautiously into any unstable region found below the maximum power level." The inspection of an AEC installation by the Division of Inspection was recommended as insurance that the routine phases of operation not be endangered because of the overriding attraction of the research program. It is also notable that the heat fluxes greater than the 50 per cent burnout limitation imposed upon the WTR will be permitted.

Heat Transfer Reactor Experiment No. 3A (HTRE-3A)

The HTRE[14] is a heterogeneous high-temperature air-cooled experimental reactor to test materials and configurations of interest to the aircraft nuclear-propulsion program. The ACRS noted that, in view of the necessity of driving the components of this experiment close to their ultimate limits, a meltdown of a fair fraction of the reactor should be considered an operational hazard and not an unlikely accident. Therefore the ACRS concluded that the experiment at the design power level and with the proposed meteorological restrictions should proceed only if a cooling-air cleaning system is provided which efficiently removes halogens and bone seekers. (W. B. Cottrell)

Hearings on Reactor Projects

The public hearing that was held during 1957 on the application by the Power Reactor Development Company, Inc. (PRDC), for a construction license for the Enrico Fermi reactor followed as a consequence of petitions filed by three AFL-CIO labor unions.* Although there were no provisions for such procedure in the Atomic Energy Act of 1954 or its amendments up to that time, the AEC ordered a hearing held. The results of this first hearing are contained in Docket F-16 entitled "Matter of the Power Reactor Development Company."†

Since the first hearing, the 85th Congress has further amended the Atomic Energy Act of 1954 by Public Law 85-256, which specifies in part that "the Commission shall hold a hearing after a 30-day notice and publication once in the *Federal Register* on each application under Section 103 or 104b for license for a facility and on any application under Section 104c for a license for a testing facility." The amendment, of which this is a part, was approved Sept. 2, 1957. The Atomic Energy Act of 1954 provided that any proceeding of this kind be "subject to judicial review in the manner prescribed" in the Administrative Procedures Act as amended. Thus the hearing process has become an authorized step in reactor licensing, and hence in the hazards-evaluation procedure.

To date, several reactor facilities have been subject to this hearing process, including those of the PRDC (Enrico Fermi), the Westinghouse Electric Corporation (Westinghouse Testing Reactor), and the Commonwealth Edison Company (Dresden Nuclear Power Station). Hearings have also been held on two of the government-owned power reactors that are to be built under the

*United Automobile, Aircraft, and Agricultural Workers of America, United Paper Workers of America, and International Union of Electrical Radio and Machine Workers.

†This and other dockets may be examined at the AEC Law Library in Washington, D. C.

Power Demonstration Program, i.e., the boiling-water reactor to be built by Rural Cooperative Power Association at Elk River, Minn., and the organic-moderated reactor to be built by the City of Piqua, Ohio. Although these reactors will not be licensed since they are government-owned, hearings are required in line with AEC policy for safety review of these reactors. A number of other reactors, principally research and training reactors, have been licensed since the hearing procedure was established; however, since no request was received from either the licensee or an intervenor within the prescribed times, the hearings in these instances were essentially a formality.

The procedure that is followed in the conduct of these hearings is described in the *Code of Federal Regulations*.[15] These rules describe the qualifications of the hearing examiner, his powers, method of intervening, conduct of a hearing, intermediate decisions and exceptions thereto, and final decisions and petitions therefrom. These rules are in keeping with the legal traditions in this country and provide the opportunity for any party to have a public hearing of actual or imagined grievances. As established in these procedures, the intermediate decision is made by the hearing examiners, and the final decision is made by the AEC on the basis of the record and any exceptions to the intermediate decision that may have been filed.

The recent injection of the hearing process into the licensing procedures has raised several issues of vital importance. These fall into two categories: (1) those having to do with the safety of the reactor facilities, as discussed below, and (2) those concerned with the mechanics of the licensing process. The dual role which the AEC plays in the nuclear power industry of both promoting the reactor facilities and regulating them has been previously criticized, but we can discern no apparent effect on the hearings to date. The public hearing is a historically valuable step in our legal process, serving as the outlet for any potentially aggrieved party. There is also every reason to expect that they will become less prolonged as the licensing branch and the hearing examiner come to understand each other's requirements for the operation of nuclear facilities without undue risk to the public. In general, however, it would also appear that, if there is no intervenor whose objections must be heard and resolved, the hearing process can add little to the safety of a reactor facility which had previously been found to be acceptable to the AEC Hazards Evaluation Branch (HEB) and to the ACRS.

Speaking before the Annual Convention of the Federal Bar Association on the subject "Industry Looks at AEC Regulations," D. C. Allen, counsel, Yankee Atomic Electric Company, stated[16] that "safety is a matter of judgment of the future, not a fact or past event that is susceptible of a yes-or-no finding." He also suggested that the hearing examiner be provided with a permanent staff in order that his decision be reached "in the light of the broadest possible knowledge of this field."

It is, however, in no sense apparent that a third technical hazard evaluation group (in addition to the HEB and ACRS) could expedite the licensing procedure, particularly since the existing technical experts can be made available to testify at the hearing examination. On the other hand, if the hearing examiner were to accumulate a backlog of cases, the appointment of one or more alternate examiners would relieve the situation.

In the appraisal of the safety of a reactor installation, whether by the ACRS, HEB, or the Hearing Examiner, the most serious deterrent to a consistent objective evaluation is the lack of accepted quantitative safety criteria and standards. The need for such safety criteria is now generally recognized within the AEC, and an extensive effort on the part of several groups both within and without the AEC is being undertaken to establish definitive criteria. As such information is developed it will be included in *Nuclear Safety*. This task is extremely difficult and has many far-reaching ramifications. The urgency of this undertaking is obvious when one considers the investment in a nuclear facility, the potential hazard, and the fact that in lieu of such information reactors continue to be designed and built with no definitive criteria other than that which other reactors have been able to "get by the Safeguards Committee with."

Power Reactor Development Company, Inc. (Enrico Fermi Reactor)

Of the cases that have been heard to date, that of the PRDC was the first and thus established the precedent for those which follow. The final decision on this case was rendered May 26, 1959, and is contained in Docket F-16, "Matter of the Power Reactor Development Company."

Some of the important consequences of this case, summarized in an article by Morrison and Garrick,[17] are as follows:

1. The hearing demonstrated that reactor safety issues can be handled by traditional administrative processes.
2. It assured public confidence in AEC's safety controls.
3. It confirmed the validity of AEC's construction permit procedures.
4. It contributed to reactor hazards evaluation.
5. It contributed to technical information concerning fast-reactor safety.

Although the hearing was concluded as far as the AEC was concerned, the intervening labor unions have brought the matter before the U. S. Court of Appeals for the District of Columbia circuit,[18] claiming that the AEC had violated its own regulations and that the unions had been denied a fair hearing.

Commonwealth Edison Company (Dresden Reactor)

The matter of the Commonwealth Edison Company has not yet been completely resolved, although the hearing has been held at frequently scheduled intervals, starting on July 9, 1959.* This hearing is somewhat different from the PRDC case in that it concerns the issuance of an operating license rather than a construction license; and, furthermore, no petition for leave to intervene was filed with the AEC. Rather, the hearing examiner, Samuel W. Jensch, was not convinced that the stipulations in the proposed operating license agreement for the Commonwealth Edison Company were sufficient to give reasonable assurance that the Dresden boiling-water reactor will operate without undue hazard to the health and safety of the public.

Some of the specific issues that have evolved out of this hearing to date are the following:

1. Should the company be required to go to the AEC for approval of *any* change in the facility design or operating procedure?
2. Should the company be required to establish an internal safety review committee (as recommended by ACRS)?
3. Must the safety of the reactor be demonstrated prior to granting an operating license, or are assurances of "no undue risk" sufficient?

To date the hearing examiner has issued two decisions that resolve these questions so far as his part in the licensing procedures is concerned. The first intermediate decision, rendered Sept. 26, 1959, permitted operation at 1 Mw for 45 days while the hearing continued; the supplemental intermediate decision, rendered Nov. 3, 1959, ordered that a license be issued for operation at 50 per cent power and described the requirements for issuance of a license to cover operation at full power.

In this supplemental decision, the hearing examiner resolved the above issues as follows:

1. "Commission approval for all changes to the enumerated specification for operation" is required. However, this does not require AEC approval of "all" changes, as had been considered, but only those enumerated in the license which would have some bearing on the safety of the reactor.
2. The hearing examiner rejected the request that the company be required to establish its own safety review committee on the grounds that the appraisal of safety is an AEC responsibility and that the organization of the company to operate the facility is an internal matter.
3. "The need for an extent of Commission approval for certain phases of the operation depends" not on assurances but "upon satisfactory demonstration shown in successive testing and operating steps." Thus, in the expectation that all fundamental performance characteristics of this large core in the Dresden reactor will have been determined by operation at an intermediate power level, the hearing examiner authorized operation at 50 per cent power.

In the resolution of this last point, it seems apparent that the hearing examiner had decided that the "reasonable assurances" which were applied in the PRDC case in the issuance of a construction license afforded inadequate grounds upon which to issue an operating license. Nevertheless, the extent to which operation of a reactor at any given power level may demonstrate

*The various releases concerning the licensing of the Commonwealth Edison Company Dresden Nuclear Power Plant are available in the Public Document Room, AEC Administration Building, Washington, D. C. Summaries of this information are listed in the issues of *Atomic Energy Clearing House* covering the period of the hearing, namely, Vol. 5, No. 37, through current issues.

the safety of operation at a higher power level is not certain.

Other questions raised in the course of this hearing examination were: (1) the role of the inspection division during the test program, (2) the surveillance role of the AEC for safety after start-up, (3) whether technical data supporting the reactor design and submitted by a licensee as "business confidential" should be withheld from the public record even though presented at an official hearing, and (4) opportunity for a hearing on amendments to the operating license.

City of Piqua Reactor

A public hearing[19] was held on Nov. 10, 1959, to consider the following points with regard to the organic-moderated reactor proposed for construction at the City of Piqua, Ohio: (1) whether there is information sufficient to provide reasonable assurance that a nuclear reactor of the general type planned for construction at the site can be constructed and operated at the proposed site without undue risk to the health and safety of the public; (2) whether there is reasonable assurance that technical information omitted from, and required to complete, the preliminary safeguards report will be supplied; (3) whether the reactor designer is technically qualified to design and construct the reactor; and (4) whether construction of the reactor will be inimical to the common defense and security or to the health and safety of the public.

At the hearing for the Piqua project, no objections to the project were voiced, and it was therefore recommended that AEC authorize Atomics International to proceed with construction. Approval for the operation of the facility is, of course, subject to another hearing to be held before operation begins. In testimony at the hearing, Dr. Biles of the HEB pointed out that the location of the reactor, where the close-in population density is high and where the nearby water is depended upon for an important source of community supply, makes it necessary that strict control of gaseous, liquid, and solid radioactivity be maintained. Later in the testimony, however, when the hearing examiner inquired whether the "character of the site emphasized the necessity for further safeguards," Dr. Biles stated that "unusually adequate safeguards were required instead of additional ones." In the course of the testimony, Dr. Biles made the following points in support of the adequacy of the safety of the reactor:

1. Analysis of the proposed design and operating experience with the OMRE indicate that radioactivity in gaseous effluents will be well within the requirements imposed by the character of the site.

2. Contamination of surface and ground waters from liquid effluents is extremely remote since coolant released from the primary system would solidify rapidly and hence immobilize any contained fission products.

3. Protection engineered into the facility is adequate to safeguard against any credible circumstances that could occur, including the "worst possible" situation.

Rural Cooperative Power Association (Elk River Reactor)

The issues to be resolved in connection with the Elk River boiling-water reactor were the same as those enumerated above for the City of Piqua reactor. Although the hearing[20] thus far has not brought to light any unique safety problems of concern, it has brought into sharp focus the problem of federal and state relations with respect to the regulations for approving reactor projects. In a letter from Governor Freeman of Minnesota, which was read at the hearing, the governor cited as prime concern the water supply source for the Minneapolis–St. Paul area and declared that the state did not "doubt that a safe installation can be designed, constructed, and operated," but that

> What we do want and need is the opportunity to independently assure ourselves, and thereby assure the people of Minnesota, that the proposed installation has been so designed and constructed and will be so operated. Objection is therefore made to the issuance of a construction permit in regard to the Elk River reactor without official prior or concurrent assurances that the federal government, as owner of the reactor, will comply with the laws and regulations of the State of Minnesota relative to nuclear reactors and water pollution.... I therefore suggest that compliance with our said laws and regulations be made a condition of the issuance of the construction permit.

Since the hearing was established to give interested parties an opportunity to take part in arriving at a safety determination of an AEC reactor, the AEC pointed out that the issuance of a construction permit under Part 50 of AEC

regulations was not involved and that, furthermore, the question of compliance with state laws appeared to be beyond the scope of the hearing. Although the AEC favored cooperation with the state, they indicated they were not able to accept language involving compliance with state regulations. Such AEC compliance was a matter of policy and a delicate legal question of federal and state regulations which could perhaps best be resolved by exchanges and meetings between AEC and state people.

Dr. Beck of the HEB in his testimony pointed out two proposed features of the plant which needed additional investigation (the properties and mechanical strength of the control rods and the details of the nuclear instrumentation to be used during start-up and operation of the reactor), but he stated that he believed both of these features could be satisfactorily resolved. He also testified that "our examination of the project has led us to the opinion that there is reasonable assurance that the Elk River reactor, as described in the general design plans, can be satisfactorily completed in design and constructed and operated at the proposed site without undue risk to the health and safety of the public." When he was questioned as to whether the reactor, as designed, constituted an undue hazard to the drinking water for the Minneapolis–St. Paul area, he stated that it did not.

The hearing for the Elk River reactor was not to decide whether a construction permit was to be granted. However, the questions raised at the hearing will no doubt result in a clarification of the question of state and federal regulations. As the atomic energy industry grows, increasing regulation will be imposed on the industry by the various states. Already many of the states have regulations to varying degrees,[21] and how these regulatory agencies and the federal government work together may well determine not only the rapidity with which the industry grows but also the degree of safety which is obtained.

(W. B. Cottrell and H. N. Culver)

Licensing of Research and Training Reactors

In a number of actions taken by the AEC in regard to research and training reactors (Iowa State University,[22] Texas Engineering Experiment Station,[23] University of Oklahoma,[24] and Rice Institute[25]), the granting of a license was made subject to the provision that no fuel element should be removed from, or inserted into, the reactor without prior AEC authorization in writing. Therefore, if it is the intention of the university to allow its students to perform critical experiments in the licensed facility, it is necessary first to submit complete information on such proposed experiments to the AEC.

In replying[24] to the AEC on this restriction, Swearinger of the University of Oklahoma said

> It would seem that the license amendment as proposed by the Commission imposes an unnecessary restriction on the use of the AGN-211 reactor as a means of training nuclear engineers. The experience gained by performing a critical assembly experiment is certainly an important part of the training program. We had planned, therefore, to disassemble the reactor core and reassemble it, using the standard proven techniques for critical assemblies, as a standard experiment in the reactor laboratory program. The necessity of obtaining written permission for each such repetition does not seem necessary.

He then points out that

> We believe that the Commission would not be in any better position to evaluate such an experiment than our Reactor Safety and Hazards Committee. If such a change in licensing is really necessary, it might be better to word it so that no *permanent* changes in core geometry may be made without prior Commission authorization in writing. There are many other ways that the excess reactivity can be increased in the AGN-211 other than changing fuel geometry. (An example is addition of polyethylene to the glory hole.) It seems unreasonable to decrease the training value of the reactor by eliminating one of the most valuable experiments, namely, the critical experiment, when the hazard is not essentially greater than the others which are already successfully coped with.

Although the resolution of this present restriction has not yet been made, it points out one of the basic questions that exists with regard to nuclear installations in both industrial and educational institutions. Although the AEC has the responsibility to protect the health and safety of the public, it is not clear to what extent this responsibility may be delegated to the reactor operator. In cases that have been reviewed and approved by the AEC, there have developed two forms of regulation: (1) the AEC has retained complete control over the operation of a particular reactor so that a license is granted for only a specific method of operation and (2) the AEC has delegated to the reactor opera-

tor specific areas where some limited changes may be made in either facility design or operation if such changes do not alter the conditions of the maximum hazard as approved in the safeguard reviews. An example of this latter type of regulation is represented by the conditions specified for the licensing of the WTR at Waltz Mills, Pa.[26,27] In this case Westinghouse is authorized to make changes in the "facility design, performance characteristics, and operating procedures" with only the approval of the Westinghouse Testing Reactor Safeguards Committee (WTRSC) if (1) "the WTRSC determines that the proposed changes involve hazards not greater than and not different from those analyzed in the final safety report, and do not involve a material alteration of the facility," and (2) the proposed changes do not fall into a specific category designated by AEC, which requires not only a review by the WTRSC but also a report to AEC describing the changes and the effect of such changes on the hazards. This report is then subject to approval by the AEC.

This latter type of regulation recognizes the fact that there are many changes of minor importance both in the design and in the operation of a reactor and that the operator is qualified to deal with such changes. The operation of a training reactor where part of the training is obtained by running critical experiments would involve changing the physical or nuclear characteristics of the core. Although this particular change is not allowed in the operation of the WTR without first reporting the proposed change to the AEC, if the critical experiments were outlined in the initial hazards report and limited to a defined type of experiment, the conduct of critical experiments would not increase the potential hazards. Thus it would be expected that the operator (or his local hazards-evaluation group) could provide adequate review of these experiments. Since, in general, reactors used for training purposes are operated at low power levels and are inherently safe, their use in the conduct of critical experiments should not represent an undue hazard to the general public.

(H. N. Culver)

Safeguards Reports and Selected Reading

The recently issued safeguards reports and other selected literature pertaining to hazards and safety of reactors are listed below for reference.* Because of the similarity of many reactors (in particular, the research reactors), the list is not intended to be all-inclusive.

1. J. Jacobson (general editor), Preliminary Safeguards Report for the Piqua Organic-moderated Reactor (Revised), NAA-SR-3575, Apr. 13, 1959. (Unclassified AEC report.)

2. Atomics International Staff, Supplement 1 to the Preliminary Safeguards Report for the Piqua Organic-moderated Reactor (Revised), NAA-SR-Memo-4048, July 1, 1959.

3. Northern States Power Company Staff, Supplement No. 1 to Preliminary Safeguards Report, Pathfinder Atomic Power Plant, ACNP-5905, June 26, 1959.

4. Pacific Gas and Electric Company Staff, Preliminary Hazards Summary Report, Humboldt Bay Power Plant Unit No. 3, Apr. 15, 1959.

5. Yankee Atomic Electric Company Staff, Yankee Nuclear Power Station, Technical Information and Final Hazards Summary Report, Vol. II, Sept. 15, 1959.

6. T. L. Gershun (general editor), Preliminary Safeguards Report Based on Uranium-Molybdenum Fuel for the Hallam Nuclear Power Facility, NAA-SR-3379; also Supplement I by Atomics International Staff.

7. Atomics International Staff, Additional Safeguards Evaluation for the Uranium-Molybdenum Fueled Core of the Hallam Nuclear Power Facility, NAA-SR-Memo-4067, July 7, 1959.

8. N. G. Wittenbrock et al., Plutonium Recycle Test Reactor, Final Safeguards Analysis, HW-61236, Oct. 1, 1959. (Unclassified AEC report.)

9. The Martin Company Nuclear Division, PM-1 Nuclear Power Plant, Hazards Summary Evaluation, MND-M-1853, Oct. 15, 1959. (Unclassified AEC report.)

10. G. O. Bright and J. E. Grund, SPERT II Hazards Summary Report, IDO-16491, Dec. 10, 1958. (Unclassified AEC report.)

11. Liquid Metal Fuel Reactor Experiment, Annual Technical Report, Babcock & Wilcox Company (Hazards Evaluation, pp. 23-33, 99-100, BAW-1136, Mar. 25, 1959. (Unclassified AEC report.)

*Many of these reports are not available, but they may be examined in the Public Document Room, AEC Building, Washington, D. C.

12. Safety and Fire Protection Branch, AEC, A Summary of Industrial Accidents in USAEC Facilities, TID-5360(Suppl. 2), September 1959. (Unclassified AEC report.)

13. W. E. Bost, Radioactive Decontamination, A Literature Search, TID-3535, September 1959. (Unclassified AEC report.)

14. W. E. Bost, Toxicity and Other Hazards of Beryllium and Rocket Propellants, A Literature Search, TID-3531, July 1959. (Unclassified AEC report.)

15. H. L. Falkenberry et al., A Preliminary Study of a Direct-cycle Steam-cooled Reactor for Merchant Ship Propulsion (Hazards, pp. 11-14), ORNL-2759, Sept. 23, 1959. (Unclassified AEC report.)

16. A. C. Foskett, Techniques for Handling Radioactive Materials, A Bibliography, AERE-Bib-122, June 1959. (Unclassified British report.)

17. W. McGuire and G. P. Fosher, Containment Studies of the Enrico Fermi Atomic Power Plant, May 1959.

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1. J. W. Greenwood, Contamination of the NRU Reactor in May 1958, AECL-850 (CRR-836), May 1959. (Unclassified Canadian report.)
2. P. R. Tunnicliffe, Atomic Energy of Canada Ltd., Nov. 4, 1955. (Unpublished.)
3. E. Siddall, A Study of Serviceability and Safety in the Control System of the NRU Reactor, AECL-399 (CRNE-582), Nov. 9, 1954. (Unclassified Canadian report.)
4. A. A. Jarrett (general editor), SRE Fuel Element Damage: An Interim Report, NAA-SR-4488, Nov. 30, 1959. (Unclassified AEC report.)
5. B. E. Harper, Atomics International. (Unpublished.)
6. C. Starr and R. Dickinson, *Sodium Graphite Reactors*, Addison-Wesley Publishing Company, Inc., Reading, Mass., 1958.
7. W. E. Parkins, The Sodium Reactor Experiment, *Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955*, 3: 295–321 (1955).
8. William L. Ginkel et al., Report of the Investigating Committee: Nuclear Incident at the Idaho Chemical Processing Plant, Phillips Petroleum Company and Idaho Operations Office, Nov. 16, 1959.
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11. Westinghouse Electric Corporation, Atomic Power Department, Aug. 7, 1959. (Unpublished.)
12. Westinghouse Electric Corporation, Atomic Power Department, October 1959. (Unpublished.)
13. J. M. West et al., Hazard Summary Report on the Experimental Boiling Water Reactor (EBWR), ANL-5781, November 1957. (Unclassified AEC report.)
    See also E. A. Wimunc and J. M. Harrer, Hazards Evaluation Report Associated with the Operation of EBWR at 100 Mw, ANL-5781 (Addendum), December 1959. (Unclassified AEC report.)
14. General Electric Company, Aircraft Nuclear Propulsion Department, Aug. 28, 1959. (Unpublished.)
15. Title 10, Part 2, Rules of Practice, *Code of Federal Regulations*, 1959 edition published by the Division of the Federal Register, the National Archives, Washington, D. C.
16. *Atomic Energy Clearing House Special Issue*, p. 2, Oct. 6, 1959.
17. J. L. Morrison and B. J. Garrick, What We Learned from the PRDC Case, *Nucleonics*, 17(7): 60 (July 1959).
18. *Atomic Energy Clearing House*, 5(45): 15–16 (Nov. 9, 1959).
19. *Atomic Energy Clearing House*, 5(47): (Nov. 23, 1959).
20. *Atomic Energy Clearing House*, 5(46): (Nov. 16, 1959).
21. *Nuclear Safety*, 1(2): 33 (December 1959).
22. *Atomic Energy Clearing House*, 5(40): (Oct. 5, 1959), and Docket No. 50-116.*
23. *Atomic Energy Clearing House*, 5(41): (Oct. 12, 1959), and Docket No. 50-59.*
24. *Atomic Energy Clearing House*, 5(44): (Nov. 2, 1959), and Docket No. 50-112.*
25. *Atomic Energy Clearing House*, 5(46): (Nov. 16, 1959), and Docket No. 50-114.*
26. Annex A Docket No. 50-22,* Westinghouse Electric Corporation License.
27. *Nuclear Safety*, 1(1): 45 (September 1959).

*On or before such time as the notice of the proposed issuance of a construction permit for any reactor facility is filed with the Office of the Federal Register, a Docket Number is assigned to the facility by the AEC Division of Licensing and Regulation. This Docket Number is then used to identify all reports, amendments, letters, notices, hearings, and other information associated with that facility.

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